• Volume 8C, Number 2, June 1960

BULLETIN OF THE RESEARCH COUNCIL OF ISRAEL

Section C
TECHNOLOGY

Bull. Res. Counc. of Israel. C. Techn.

Incorporating the Scientific Publications of the Technion — Israel Institute of Technology, Haifa

Page

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A MODEL FOR REPEATED RESIDUAL DEFORMATION UNDER REPEATED LOADING OF EQUAL DIRECTION

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ABSTRACT

Residual stresses in steel structures are discussed for the special case where the critical collapse load is more than double the maximum elastic load (producing yield stress).

A structural model is suggested illustrating disadvantageous repeated residual deformation under repeated loading of equal direction.

We refer to statically indeterminate structures made of ideal plastic steel. Simple beams may also be regarded as statically indeterminate (internally), insofar as the stress distribution is based on a deformation criterion (Bernoulli's hypothesis).

Such structures, when loaded into the plastic range, retain upon unloading residual deformations and stresses. Under repeated loading in the *same direction* the behaviour of the structure is entirely elastic* up to the original load level[1], the exceptional case being when opposite plastic deformations occur during unloading, so that repeated loading would cause repeated yielding endangering the structure in a manner similar to reversed loading.

This will happen when the critical (plastic) collapse load P_{cr} is more than double the maximum elastic load P_Y producing the yield stress σ_Y in the most stressed element of the structure:

$$\alpha = \frac{P_{cr}}{P_Y} > 2 \tag{1}$$

Consider, for instance, a simply supported beam of symmetrical triangular section under load P (Figure 1,*a*; *C*–centroid). Under the critical load P_{cr} the section becomes a plastic hinge[2] (being subjected to the full plastic moment M_{pl}): stresses are distributed according to Figure 1,b, and the neutral axis n_{pl} divides the section area into two equal parts. The shape factor α for a triangle being 2.3, this beam may serve as an example for the case represented by eq. (1). Upon unloading P_{cr} the response is immediately elastic, and therefore the residual stresses are obtained by superposition of the respective stress diagrams due to M_{pl} (Figure 1,b) and to a re-

* Residual stresses are thus usually of an advantageous nature, provided the loading is confined to one direction only. If reversed loading takes place, opposite yielding will set in under a smaller load than the first yield load; reversed loading, however, is not considered in this discussion.

Received March 25, 1960.

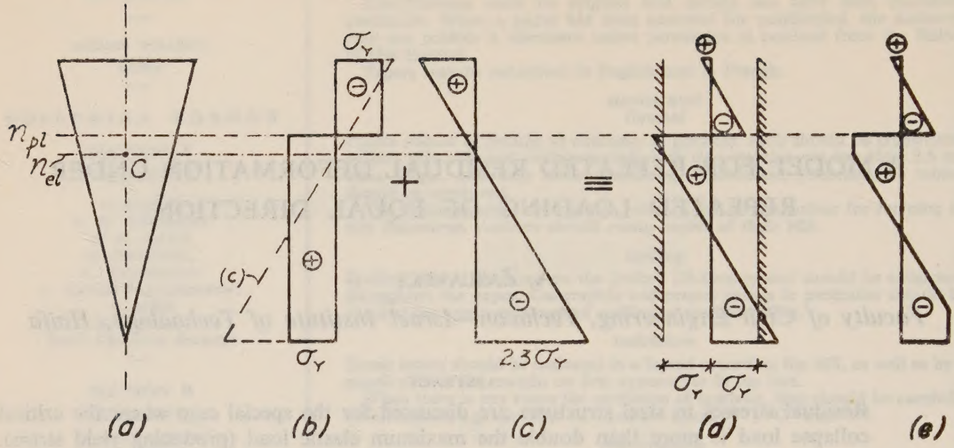

Figure 1

versed elastic moment M_{pl}^{el} (Figure 1,c) equal in magnitude to $M_{pl}^{el} = M_{pl} = \alpha M_Y = 2.3\ M_Y$ (M_Y – maximum elastic moment). This would result in the diagram shown in Figure 1,d, which is ruled out because the yield stress $\pm\sigma_Y$ cannot be exceeded. In these circumstances, yielding will occur both at the lowest fibre and near the neutral axis during the last stage of unloading.* The approximate final residual stress distribution is shown in Figure 1,e.

A simple structural model, singly indeterminate, for analysis of the case $\alpha > 2$ is suggested in Figure 2,a. A rigid beam AB is hinged at A and supported by two steel bars 1 and 2, of equal length but different sections $A_1 > A_2$. The structure supports a vertical load P at C. (No buckling assumed).

Both bars being of equal length, bar 2 will be the most stressed ($\sigma_2 = \sigma_1 b/a$) and the maximum elastic load P_Y producing $\sigma_2 = \sigma_Y$ equals:

$$cP_Y = \sigma_Y A_2 b\ (\frac{A_1}{A_2}\ \frac{a^2}{b^2} + 1).$$

The critical load P_{cr} causing the yielding of both bars equals (by $\Sigma M_A = 0$):

$$cP_{cr} = \sigma_Y A_1 a + \sigma_Y A_2 b = \sigma_Y A_2 b\ (\frac{A_1}{A_2}\ \frac{a}{b} + 1).$$

Therefore:

$$\alpha = \frac{P_{cr}}{P_Y} = \frac{\left(\dfrac{A_1}{A_2}\right)\left(\dfrac{a}{b}\right) + 1}{\left(\dfrac{A_1}{A_2}\right)\left(\dfrac{a}{b}\right)^2 + 1}. \tag{2}$$

* Yielding near the neutral axis would also occur in unsymmetrical sections (e.g., T-section) irrespective of the $a > 2$ condition.

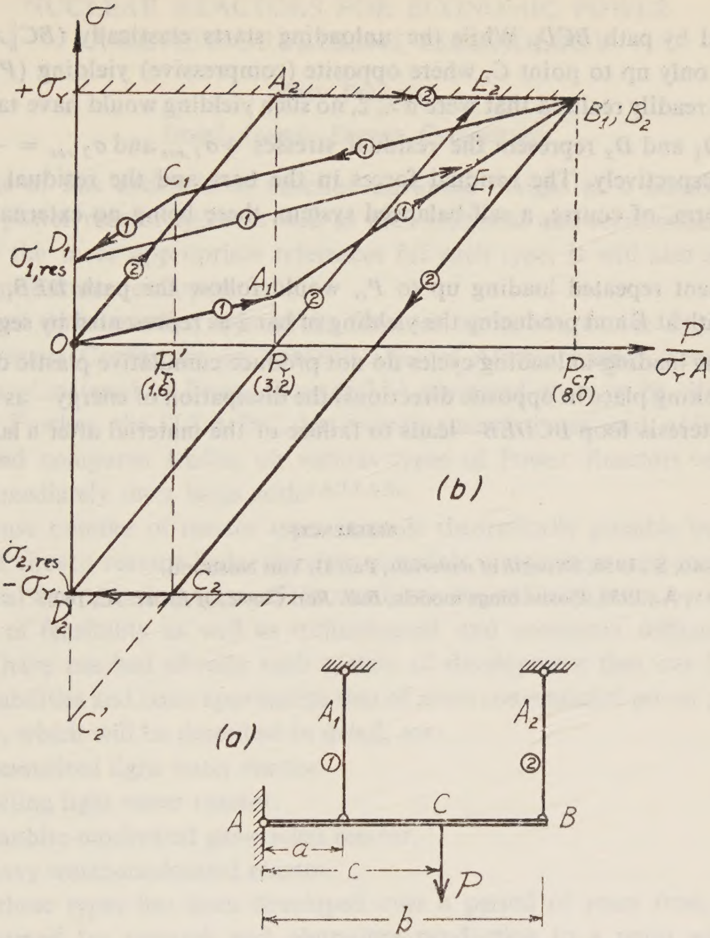

Figure 2

Assigning the model to work at $\alpha = 2.5$, eq. (2) leads to:

$$\left(\frac{a}{b}\right)^2 - 0.4\left(\frac{a}{b}\right) + 0.6\ \frac{A_2}{A_1} = 0$$

$$\frac{a}{b} = 0.2 \pm \sqrt{0.04 - 0.6\ (A_2/A_1)}$$

By choosing $A_1 = 15A_2$ we obtain a real and convenient ratio: $a/b = 0.2$. Figure 2,b shows the loading history $\sigma\Big/\frac{P}{\sigma_Y A}$ for both bars of the model, assuming: $a = 0.2\ m$; $b = 1.0\ m$; $c = 0.5$; $d = 0.5\ m$; $A_1 = A$; $A_2 = 15A$.

Path OAB (i.e., OA_1B_1 for bar 1 and OA_2B_2 for bar 2) shows the loading up to P_{cr} through P_Y ($P_{cr} = P_Y = 2.5\ P_Y = 2.5 \times 3.2\ \sigma_Y A = 8.0\ \sigma_Y A$). Unloading is

represented by path BCD. While the unloading starts elastically ($BC \| AO$) it continues so only up to point C, where opposite (compressive) yielding (P'_Y) starts in bar 2. It is readily realised that were $\alpha \leqslant 2$, no such yielding would have taken place.*

Points D_1 and D_2 represent the residual stresses $+\sigma_{1,res}$ and $\sigma_{2,res} = -\sigma_Y$ of bars 1 and 2 respectively. The residual forces in the bars and the residual reaction at hinge A form, of course, a self-balanced system, there being no external load after unloading.

Subsequent repeated loading up to P_{cr} would follow the path DEB, joining the original path at E and producing the yielding of bar 2 as represented by segment E_2B_2.

While the loading-unloading cycles do not produce cumulative plastic deformation (yielding taking place in opposite directions) the dissipation of energy—as represented by the hysteresis loop $BCDEB$—leads to failure of the material after a large number of cycles.

REFERENCES

1. TIMOSHENKO, S., 1956, *Strength of materials*, Part II, Van Nostrand.
2. ZASLAVSKY, A., 1959, Plastic hinge models, *Bull. Res. Counc. of Israel* 7C, 167.

* It may also be noted that $P'_Y = 2P_Y - P_{cr}$ and the limiting case $\alpha = 2$ may be formulated as $P'_Y = 0$ and eq.(1)—as

$$P'_Y < 0 \qquad (1,a).$$

NUCLEAR REACTORS FOR ECONOMIC POWER
A REVIEW OF PRESENT TECHNOLOGY

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Israel Atomic Energy Commission

The purpose of this article is to describe advanced designs of a number of well established power reactor types as well as their technical and economical problems and to give the most appropriate references for each type. It will also attempt to forecast some future developments.

The present time is rather appropriate for such an article, as the 2 years since the 1958 Geneva Conference on Peaceful Uses of Atomic Energy were sufficient for the mass of material on Power Reactors[1,2,3] presented there to be digested and compared[4]. Further, the U.S.A.E.C. has recently commissioned and, at least partly, published and compared studies on various types of Power Reactors which could be built immediately on a large scale[5,6,7,8,9,9a].

An immense number of reactor types is made theoretically possible by the combinations of chosen reactor fuels, cladding materials, moderators and coolants, and their physical and chemical states. This number is limited in practice by many considerations of feasibility as well as technological and economic difficulties. Only four types have reached already such a state of development that our knowledge of their capabilities and costs approaches that of more conventional power producers. These types, which will be described in detail, are:

1. The pressurized light water reactor.
2. The boiling light water reactor.
3. The graphite-moderated gas-cooled reactor.
4. The heavy water-moderated reactor.

Each of these types has been developed over a period of years from the early reactors destined for research and plutonium production to a point where large power stations either are under construction or could be started without the need for much further research and development.

THE PRESSURIZED LIGHT WATER REACTOR (P.W.R.).[10]

This type was originally developed for submarine propulsion and till now all atomic submarines and surface navy vessels have P.W.R. power plants, though these reactors, as far as can be judged from the scarce data published, are very different from economically designed plants[11].

The only working large-scale power station using P.W.R. is the Shippingport, Pa., plant in the U.S.A.[12,13,14,15,16,17,17a]. Two further large plants are under construction there[18,19,20,21,22], one in Italy[23] and in the USSR[24,25]. A demonstration power reactor is operated in the U.S.A. as by the Army prototype for a package power reactor[26] and another is nearly completed in Belgium[27].

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However, the maximum capability of the P.W. Reactors, as extrapolated from accumulated design and operation experience, is best illustrated by a design which was proposed to the U.S.A.E.C. by the Stone & Webster Engineering and Combustion Engineering Corporation.

The purpose of their report and parallel ones submitted by other groups on other reactor types[5,6,7,8,9] was to obtain data on all these types based on equal, well defined "ground rules" which would permit valid comparisons to be drawn. No undue extrapolation from available technology was permitted, as the reactors had to be designed for start of construction in 1960.

The proposed P.W.R. (Figures 1, 2) has a thermal output of 685 MW, giving an electrical output of 248 MW gross & 236 MW net. This size was found to be the maximum which could be built at present, the limiting factor being the dimension of the pressure vessel. Economic considerations would favour even larger reactors which, however, could only be built with farther advances in technology.

The reactor core consists of fuel rod clusters (Figure 3) arranged in a shape approximating a cylinder, 2.40 m dia and 2.85 m high. The core is divided into an inner zone consisting of uranium enriched to 2.6% with U_{235} and an outer one with 3.4% enrichment. Altogether the core contains 56 tons of UO_2 in the form of pellets, 1 cm dia. and 1 cm high, contained in stainless steel tubes 305 cm long. To increase the core lifetime between refuelling operations to 3½ years, the initial reactivity is reduced by including in the fuel section boron "burnable poison"[28a]. Figure 4

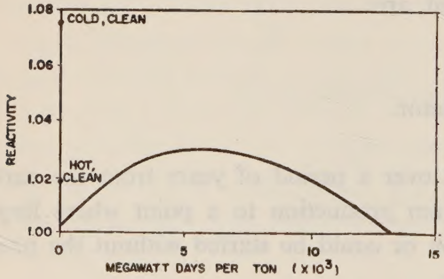


Figure 4.

Reactivity variation with burnup — two zone core at design power and temperature ADVANCED PRESSURIZED WATER REACTOR

shows the reactivity of the core varies during its life; the burnup attainable from reactivity considerations is 13,000 MW days/ton average. The designers claim that this burnup, which means up to 25,000 MW days/ton for the highest flux fuel elements, is feasible even taking into account the extensive radiation damage caused to the fuel elements.

In spite of the burnable poison, the clean, cold core has 9.4% excess reactivity. This would be difficult to control with the normal types of absorbing control rods, and a "neutron rectifier" rod with fuel section (shown in Figure 3) is proposed which gives increased control rod efficiency by simple means. The rod consist of a normal fuel cluster surmounted by a hollow neutron absorber section of silver–indium–cadmium alloy clad in stainless steel which enters the core when the rod is

lowered. This hexagonal absorber section is filled with cooling water, which increases the absorption effect by efficiently thermalizing epithermal neutrons, of which such a core contains up to 10%. It is claimed that rod effectiveness is increased 2.5 times by this means. This control rod weighs over 1 ton and requires a special drive mechanism.

The relatively high thermal efficiency of 34.5% is achieved by means of the steam circuit shown in Figure 5, which includes reheating the steam between turbine sections by fresh steam, and by the high average temperature of the pressurized water flowing through the core – 318°C at a flow rate of 20,000 tons/hours; this arrangement enables the four heat exchangers to raise steam at 70 at. pressure.

The coolant pressure of 135 at. is limited by the pressure vessel dimensions, in particular the thickness (23 cm) of the steel (Figure 1). To obtain the full-load coolant exit-temperature of 328°C, bulk boiling is allowed in the hotter core regions while a sufficient burnout safety margin is retained by keeping the ratio of burnout heat flux to peak core heat flux at 2.9.

The 4 heat exchangers and 4 main coolant pumps with their associated valves and piping are designed for economy and minimum space inside the containment sphere; in particular, expensive canned-rotor pumps are replaced by controlled-gland-leakage regular pumps. Figure 6 shows the arrangement of reactor, shield, steam generators and crane inside the 41 m dia. vapour containment sphere made of 2.8 cm thick steel. It also shows the shield tank above the reactor, which is filled with water during refuelling, so that fuel elements can be withdrawn from the open pressure vessel under water and transferred through the chute (Figure 6) to a spent fuel pool outside the sphere. The pressurizer, also shown there, is heated by 170 electric heaters and is connected to the main coolant circuit in which it maintains nearly constant pressure under transient and steady load changes from start-up to overload.

The reactor pressure vessel is surrounded by a water tank for neutron shielding, which in turn is surrounded by the primary shield of reinforced concrete, designed to make the portion lying outside accessible while the reactor is shut down.

In conjunction with the secondary shield lining the lower half of the sphere, it attenuates radiation of the operating reactor and the coolant circuit to within tolerable limits outside the sphere. Each main coolant loop has its additional shield to protect maintenance personnel. A demineralizer is provided within the sphere for purification of the coolant; this also has its own shielded compartment.

The turbo-alterator is designed for semi-outdoor operation; for the detailed design of this and the rest of the plant see ref. (6).

Of all the U.S. developed reactors, the P.W.R. has received the most engineering effort, and we can assume that present design and construction capability closely approach the ultimate development of this type. The scope for further advance in the stationary power field is therefore limited.

BOILING LIGHT WATER REACTOR (B.W.R.)[29,30,31]

The usual distinction between pressurized and boiling water reactors tends to become obsolete as far as conditions in the core are concerned. We have seen that the advanced P.W.R. permits bulk boiling in parts of the core. The main distinction now is the flow of at least part of the steam directly from the reactor to the turbine, which only the B.W.R. permits.

Again, there is only one major power station using this type reactor, in Dresden, Ill., U.S.A., which has recently gone critical (six months ahead of schedule) and is now in the "debugging" stage[32,33,34]. This was preceded by two experimental prototypes, the EBWR at the Argonne National Laboratory[35,36,37,38,26] which pioneered the type. The prototype in turn was preceded and accompanied by a number of experimental boiling reactors[39,39a] and the General Electric Vallecitos reactor[40,41].

Plants under construction include two in the U.S.A.[42,43] one in the USSR[24], one each in Italy[44,45] and Japan[46], and a demonstration plant in Germany.

However, we will only describe the most advanced technology, following the report submitted to the U.S.A.E.C. by a planning group of Ebasco Services and the General Electric Co.[5,47].

The designers examined four different B.W.R. steam circuit concepts (Figure 7) and various reactor sizes; they concluded that with present technology the optimum would be a dual-cycle forced circulation plant of 980 MW thermal output and electrical output 320 MW gross and 306 MW net. Both smaller and slightly larger plants are found more expensive, the limiting factor being availability of single shaft turbine-generators. Details of the optimum plant are given in the flow diagram, Figure 8. The design follows rather closely that of the Dresden Station; details of the reactor core and vessel are shown in Figure 9, and the arrangement of the contained components is pictured in Figure 10.

The reactor is fuelled by 63 tons of UO_2 enriched to 1.91%; however, in this case, the cladding has to be Zircalloy and the core contains considerably more moderator than in the P.W.R. (moderator to fuel ratio 2.5, as against 1.12 in the P.W.R. proposal). The latter feature, which derives from the consideration of the steam void coefficient of reactivity that determines the stability of operation, makes for a larger core and pressure vessel diameter. As the possibility of vessel fabrication is a limiting feature, lower steam conditions (65 atm) must be accepted, and the thermal efficiency of the steam cycle is 31.2% only. The control rods are the regular absorber type of cruciform shape, as shown in figure 11.

The development capabilities of B.W.R. are considerable and a quotation from (5) shows where improvements are expected:

A. *In capital cost area*

1. Increase of core specific power by approximately a factor of two.

2. Mechanical simplification of reactors, systems, and supporting services and facilities, resulting in more compact plants.

3. Increase of practical unit sizes to the range of 400–500 MW (e).

4. Improvements of steam conditions so that temperatures and pressures are comparable to those of modern, conventional plants.

Increase of the thermal efficiency from 28 to 35%.

B. *In fuel cost area*

The most important factors which will improve fuel performance and decrease costs are:

1. Decrease of present fabrication cost of fuel.
2. Increase of exposure lifetime.
3. Increase of conversion ratios.
4. Decrease of plant heat rates.

"The most important advance would be nuclear superheating, which is at present being explored by several groups"[50,51,52,53].

THE GRAPHITED MODERATED GAS COOLED REACTOR

This type, which was pioneered in England[54,55,56] and France[57,58,59] has now also attracted much attention in the USA, though there the tendency is to use enriched rather than natural uranium fuel so as to shift the cost from capital to operating expenses, which is desirable in countries with high interest rates, such as in the USA.

Two U.S. study groups on this type of reactor have recently published their findings: Oak Ridge[60,61] and Kaiser-ACF[63-67]. However, present technology will be described here by reference to the Hinkley Point power station in England, which is the most modern station for which detailed design data are available[68] though recently an outline of the latest British station, Trawsfynydd, was published[69].

Besides the 5 British[68,69,70,71,72], and 2 French stations[59] there is one such station under construction in Italy[73] and one in Japan[74,76].

The Hinkley Point station (Figures 12, 13, 14) consists of two reactors, each with a net electrical output of 250 MW from a thermal output of 966 MW, i.e. a thermal efficiency of 26%. This is achieved, in spite of the considerable load demand of the gas circulators (25 MW) by the use of a dual steam circuit, generating steam at two pressures rather as in the dual-cycle B.W.R.[75,75a]. The temperatures and flow quantities are shown in Table I. Each reactor contains 250 tons of metallic natural uranium, clad with magnesium alloy.

An important advantage of this reactor type is the possibility of unloading and reloading fuel while the reactor is in operation, a practical necessity due to the shorter reactivity lifetime of natural uranium fuel. Large and heavy fuel handling machines (Figure 15) are needed for this purpose.

Possibilities of further technical advances of this type are bound up with increasing gas temperatures beyond the present limits of temperatures permissible for fuel and canning, and much research work has been done toward the building of prototype high-temperature gas-cooled reactors in both England and the U.S.A.[76,77,78,-]

[79,80,81,81a]. A prototype is under construction in England, with many European countries participating (project Dragon[78]), and an advanced gas-cooled reactor system of 28 MW electrical output for testing new types of fuel elements is also under construction[78a].

TABLE I

Hinkley point power station data

a) Boilers and turbines	
No. of steam rising units	12
HP steam pressure at superheater outlet	640 psig
HP steam temperature at superheater outlet	685°F
LP steam pressure at superheater outlet	170 psig
LP steam temperature at superheater outlet	660°F
Total Steam flow from S.R.U.s/reactor	2.76×10^6 lb/hr
Total HP steam flow from S.R.U.s/reactor	1.76×10^6 lb/hr
Total LP steam flow from S.R.U.s/reactor	1.00×10^6 lb/hr
No. of main turbines	6
Output of each turbine	87 Mw
No. of variable frequency turbines	3
Load on variable frequency turbine/reactor	25.2 Mw
Heat transferred to gas by circulators/reactor	22.9 Mw
Maximum capability of each V.F. turbine	33 Mw
Heat transferred from gas in boiler/reactor	986.2 Mw
b) Main circulators	
No. of circulators	12
Type of circulator	single stage axial
Blower temperature rise	5.0°C.

HEAVY WATER MODERATED REACTORS[82–86]

The D_2O moderated reactor shares with the graphite moderated one the possibility of working with natural uranium and needs considerably less fuel than the latter.

Nevertheless, no large scale power station of this type has yet been started, one prototype being under construction in Canada[87] and one, for combined power and heat output, in Sweden[88]. A boiling heavy water experimental reactor is in operation in Norway[89,90].

However, present possibilities will again be described by references to a U.S. study[8,91] made by Sargent and Lundy in cooperation with the Argonne National Laboratories. This center was designed with the condition of being able to operate on natural fuel, though economic advantage might dictate the use of enriched fuel in the same reactor under U.S. conditions.

This design, far less detailed than those in the parallel studies on P.W.R. and BWR, is for a boiling D_2O pressure tube type, with heavy steam directly driving the turbo-generator having a 208 MW net output.

As shown in Figures 16, 17, 18, the reactor consists of an aluminium "calandria" vessel, 5.50 m in diameter and 6.10 m high, which contains the cold D_2O moderator and through which 287 aluminium tubes pass. 268 of these contain Zirconium pressure tubes inside which the fuel elements are mounted.

Sub-cooled D_2O enters the pressure tubes at the lower end, is heated in the fuel elements and leaves the upper end as 75% steam by volume. The steam from all tubes is collected in large steam separator drums and goes directly to the turbine at a pressure of 52 at., while the liquid coolant is recirculated through the pressure tubes. Condensed D_2O is returned to the reactor and mixed with the recirculated heavy water from the steam drums.

This design was chosen as the best of several possibilities, including pressurized and boiling pressure vessel designs and pressurized, boiling, organic and gas cooled pressure tube designs. It is believed that with natural uranium a burnup of 4,600 MW day/ton could be achieved. A Canadian proposal speaks of 8,000 MW–day per ton[92].

OTHER TYPES

It is not possible to mention here all the many other feasible power reactor types, in particular the fast and thermal breeder reactors, which, while posing onerous technical problems, promise in the end the best fuel utilization and cheapest power.

Reactors using organic liquids instead of water for moderation and cooling, have also been extensively studied[7] but no major station of this type is yet envisaged.

ECONOMICS

The United States Atomic Energy Commission has published[9] a preliminary evaluation of the status reports of the various types of power reactors, adjusting the costs to a common base, and adding the results of reasonably foreseeable advances for each type which would bring down its costs. The costs of each type of reactor, if built to-day are shown in Table II[9a]; reasonably expected improvements during the next ten years should bring these services down to the values shown in Table III[9b].

The studies, by means of which the various status reports were normalized to common conditions of financing, siting, construction and operation were carried out by Sargent & Lundy together with the Argonne National Laboratory and have recently been published[9a]. These studies included investigations on the influence of station size on capital costs as well as estimates of the operation and maintenance costs for each type. The resultant normalized power costs are given in the last column of Table II, which details the various components making up the total power cost, and compares them with the most modern conventional plant for two different values of conventional fuel price.

Figure 19 presents the information of Table II in the form of curves. It must be emphasized that neither table nor curve purports to give actual cost data for a specific station, but rather the influence of reactor type and size on costs; in particular, the single line curves of Figure 19 are not warranted by experience. However, the normalization[9a] is most valuable even with the above reservation, since this is the first time that such extensive and soundly based comparison of various reactor types has been made possible.

TABLE II

Summary of normalized power generation costs

Reactor type	Plant capacity MW(e)	Capital costs: Total capital cost $	Capital costs: Ann. costs $/yr.	Capital costs: Mills/kwh	Fuel costs: Ann. costs $/yr.	Fuel costs: Mills/kwh	Oper. & Maint. costs: Ann. costs $/yr.	Oper. & Maint. costs: Mills/kwh	Nuclear Insurance costs: Ann. costs $/yr.	Nuclear Insurance costs: Mills/kwh	Total power costs: Ann. costs $/yr.	Total power costs: Mills/kwh
Thermal Converter Reactors												
Pressurized water	75	32,600,000	4,560,000	8.69	2,405,000	4.58	719,050	1.37	370,600	0.70	8,054,850	15.34
	200	56,400,000	7,900,000	5.64	5,490,000	3.92	1,091,300	0.78	481,700	0.34	14,963,000	10.68
	300	73,400,000	10,300,000	4.90	8,000,000	3.81	1,240,000	0.59	552,900	0.26	20,093,200	9.56
Boiling water	75	35,200,000	4,930,000	9.40	2,190,000	4.17	728,050	1.38	369,000	0.71	8,217,050	15.66
	200	62,200,000	8,710,000	6.22	5,350,000	3.82	1,111,300	0.79	498,400	0.36	15,669,700	11.19
	300	78,900,000	11,050,000	5.26	7,300,000	3.47	1,270,300	0.61	565,250	0.27	20,185,550	9.61
Organic cooled	75	26,200,000	3,670,000	6.98	3,320,000	6.33	985,800	1.88	336,400	0.64	8,312,200	15.83
	200	48,200,000	6,740,000	4.81	8,250,000	5.90	1,800,300	1.28	448,500	0.32	17,238,800	12.31
	300	66,000,000	9,225,000	4.39	12,020,000	5.72	2,293,300	1.09	525,600	0.25	24,063,900	11.45
Sodium graphite	75	42,500,000	5,950,000	11.30	4,653,000	8.85	788,800	1.50	387,950	0.74	11,779,750	22.39
	200	72,000,000	10,090,000	7.20	11,174,000	7.97	1,251,300	0.89	531,500	0.38	23,046,800	16.44
	300	90,900,000	12,710,000	6.05	16,146,000	7.68	1,471,300	0.70	607,400	0.29	30,934,700	14.72

Breeder Reactors												
Fast, sodium cooled	75	34,100,000	4,770,000	9.10	4,150,000	7.90	849,550	1.61	351,950	0.67	10,121,500	19.28
	150	51,000,000	7,140,000	6.80	7,900,000	7.52	1,240,300	1.18	451,700	0.43	16,732,000	15.93
	200	60,000,000	8,400,000	6.00	10,370,000	7.40	1,391,300	0.99	488,950	0.35	20,650,250	14.74
	300	76,500,000	10,700,000	5.10	14,930,000	7.10	1,673,300	0.79	554,150	0.26	27,857,450	13.25
Thermal (Aqueous homogeneous	75	33,800,000	4,686,000	8.92	1,340,000	2.55	1,937,050	3.69	410,200	0.70	8,373,250	15.86
	200	72,300,000	10,000,000	7.15	3,010,000	2.15	3,907,300	2.79	535,850	0.38	17,453,150	12.47
	300	96,900,000	13,400,000	6.38	4,460,000	2.12	5,302,300	2.53	633,350	0.30	23,795,650	11.33
Natural Uranium & Recycle Reactors												
Heavy water	75	48,000,000	6,620,000	12.60	2,420,000	4.61	956,300	1.82	426,500	0.81	10,422,800	19.84
	200	85,000,000	11,675,000	8.35	6,170,000	4.40	1,594,300	1.14	583,300	0.42	20,022,600	14.31
	300	108,000,000	14,800,000	7.05	8,870,000	4.22	1,924,300	0.91	676,700	0.32	26,271,000	12.50
Gas cooled	75	50,600,000	7,090,000	13.50	2,020,000	3.85	728,050	1.38	432,900	0.82	10,270,950	19.55
	200	90,500,000	12,670,000	9.05	4,910,000	3.50	1,111,300	0.79	600,650	0.43	19,291,950	13.77
	300	114,000,000	15,960,000	7.60	7,040,000	3.35	1,270,300	0.61	696,500	0.33	24,966,800	11.89
Conventional Coal Fired Plants												
35c/million Btu fuel	60	13,254,00	1,855,600	4.41	1,628,700	3.88	317,200	0.75			3,801,500	9.04
	200	35,690,000	4,996,600	3.57	4,684,800	3.34	707,200	0.50	—	—	10,388,600	7.41
	325	53,795,000	7,531,300	3.31	7,549,900	3.32	832,000	0.36			15,913,200	6.99
25c/million Btu fuel	60	13,254,000	1,855,600	4.41	1,163,300	2.77	317,200	0.75			3,801,500	7.93
	200	35,690,000	4,996,600	3.57	3,346,300	2.39	707,200	0.50	—	—	9,050,100	6.46
	325	53,795,000	7,531,300	3.31	5,392,800	2.37	832,000	0.36			13,756,100	6.04

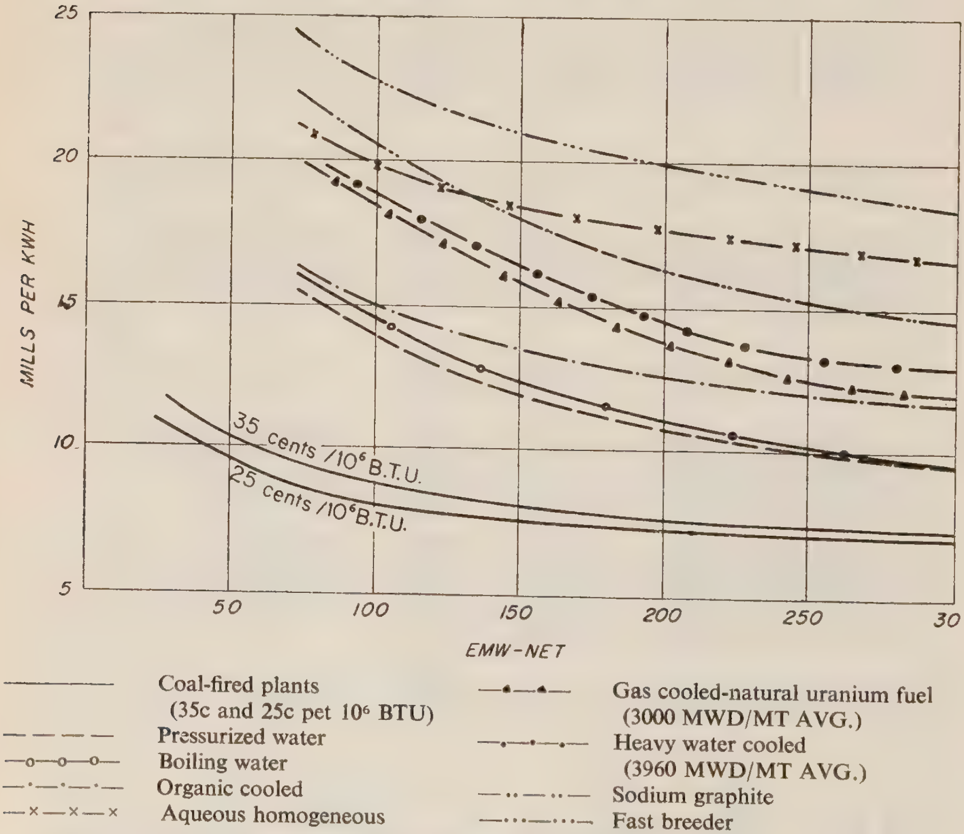
Note: Nuclear costs based on 80% Load Factor, 14% Fixed Charges, 4% Uranium Use Charge and $12.00/gm. Plutonium Credit.
Coal Fired costs based on 80% Load Factor and 14% Fixed Charges.

TABLE III
Power generation costs with improvements expected to be realized by 1970

Reactor type	PWR	BWR	OMR	SGR	GCR	D20	FBR
Total capital cost ($10^6)	64.0	64.5 (58.5)	53.2	67.1	69.5	88.6	65.0
Power costs (mills/kwh)							
Capital charges	4.40	4.31 (3.91)	3.53	4.47	4.63	5.80	4.43
Fuel cycle	2.56	2.29 (1.96)	1.83	2.00	2.62	1.21	1.99
Operation and maintenance	0.59	0.61 (0.61)	1.09	0.70	0.49	0.91	0.79
Nuclear insurance	0.25	0.24 (0.23)	0.22	0.25	0.24	0.28	0.25
Total	7.80	7.45 (6.71)	6.67	7.42	7.98	8.20	7.46

Reactor Types:
PWR — pressurised water reactor
BWR — boiling water reactor (figures in brackets assume feasibility of nuclear superheat)
OMR — organic moderated reactor
SGR — sodium graphite reactor
GCR — gas cooled reactor (graphite)
D20 — heavy water moderated reactor
FBR — fast breeder reactor

Figure 19.
Power generation costs for conventional and nuclear plants



The U.S.A.E.C. is just now (April 1960) issuing the complete results of the evaluation, and Table III is based on preliminary reports of that evaluation[9a].

Generating costs from the most modern coal burning plants in the U.S.A. are expected to vary from 6–8 mills/kwh during the next ten years, depending on coal cost in the area considered.

The tables show therefore that 1) no natural uranium fuelled reactor is expected

to be competitive with coal burners in the U.S.A. in the next decade; 2) if the considerable cost reductions shown in Table II, especially in fuel costs, can be realized, most of the systems surveyed will be competitive or marginally competitive by 1968, the exception being the D_2O natural uranium reactor and the thermal breeder.

Naturally the cost figures apply only under U.S. conditions (with 14% annual capital charges), and under different conditions quite a different order of merit will be obtained.

In particular the D_2O reactor will appear much more favourable in low capital charge countries. Under British conditions, generating costs for the most modern coal fired stations vary from 0.5–0.65 d./kwh so that the nuclear station of latest design, which deliver electricity of 0.65–0.7d./kwh, will be just competitive with the "away from coalfield" stations.

ENRICHMENT WITH PLUTONIUM

Under U.S. conditions, both graphite and D_2O moderated reactors show cost advantages when designed for enriched fuel.

Many calculations have been made[93,94,95,96] to find out whether use of plutonium would confer this advantage also on reactors which cannot be supplied with enriched uranium, such as those in countries without diffusion plants and wanting to rely only on fuel produced by themselves. Furthermore, these calculations include comparisons of the values of plutonium and U-235 as fuels. It is recalled that every reactor based on natural uranium is producing plutonium.

Plutonium may be used in a variety of reactor types, not necessarily of the same type in which it was produced.

The following uses can be envisaged[93]:

1. Mixed with natural uranium to obtain a smaller reactor (instead of U-235 enrichment).
2. Mixed with natural uranium to obtain higher effective burnup (also instead of U-235 enrichment).
3. Alone in special purpose reactors (fully enriched fuel).
4. In a breeder (mixed with depleted or natural uranium).
5. As seed elements in a seed and blanket arrangement.

Uses 1 and 2 include "spikes" of plutonium elements among the natural U elements, as well as homogenous Pu–U mixtures; in each case the economic advantage of either possibility must be investigated.

Plutonium recycling or the use of plutonium extracted from one fuel charge to enrich the next charge of the same reactor is economically worthwhile only when the plutonium cannot be used to greater advantage in any of the other mentioned schemes. This and the possible military demand for plutonium determine the credit to be assumed per gram of plutonium extracted from the fuel; once this price has been set, the validity of the recycling scheme can be calculated[96a].

To investigate the technical feasibility of plutonium recycling, a Plutonium Recycle Test Reactor[97,98,99,100] is now under construction at Hanford. This facility includes a light water-moderated pressure-tube experimental reactor and a pluton-

ium fuel fabrication plant. Experiments with research reactors such as the M.T.R. have already shown the feasibility of plutonium fueling.

No detailed design or calculations for plutonium enriched large power reactors are yet published, but it can be expected that the relative economic merit of several reactor types described in this article will be appreciably changed when this possibility is accepted.

ACKNOWLEDGEMENT

Thanks are due to the Israel Atomic Energy Commision for permission to publish this article.

Particular gratitude is expressed to Miss Alina Gralewska and Miss Sylvia Scapa for their bibliographical and editorial assistance.

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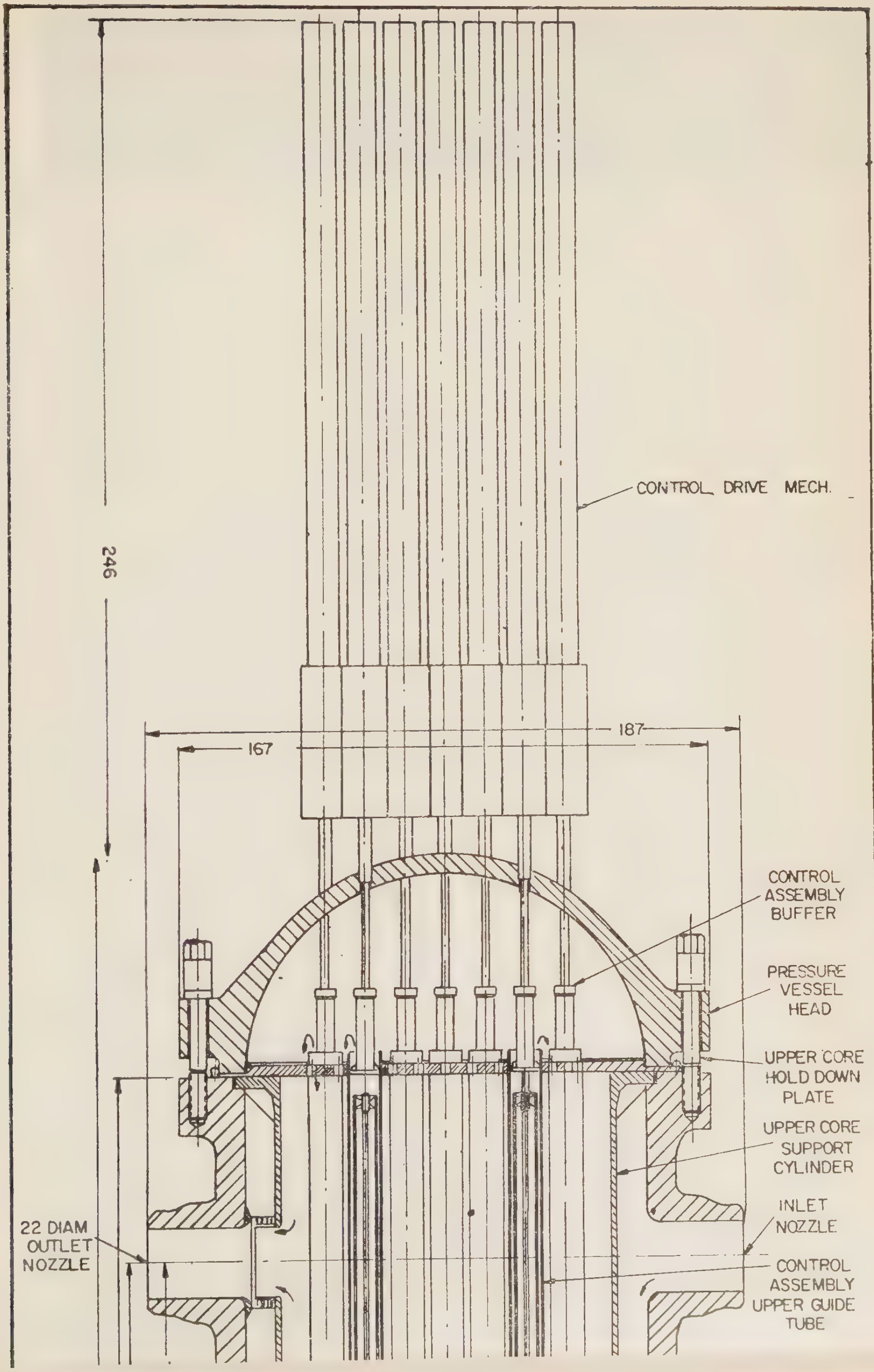
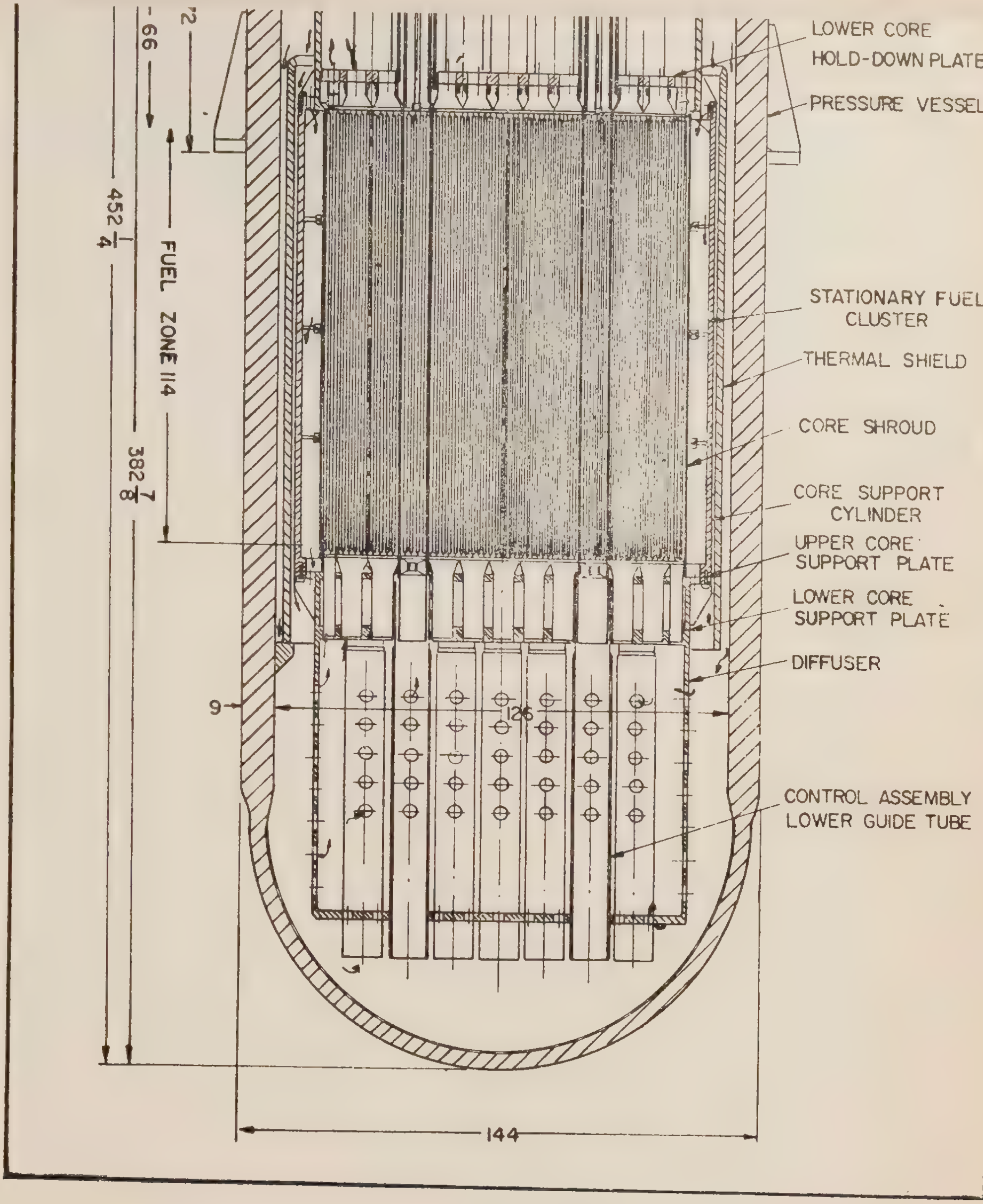
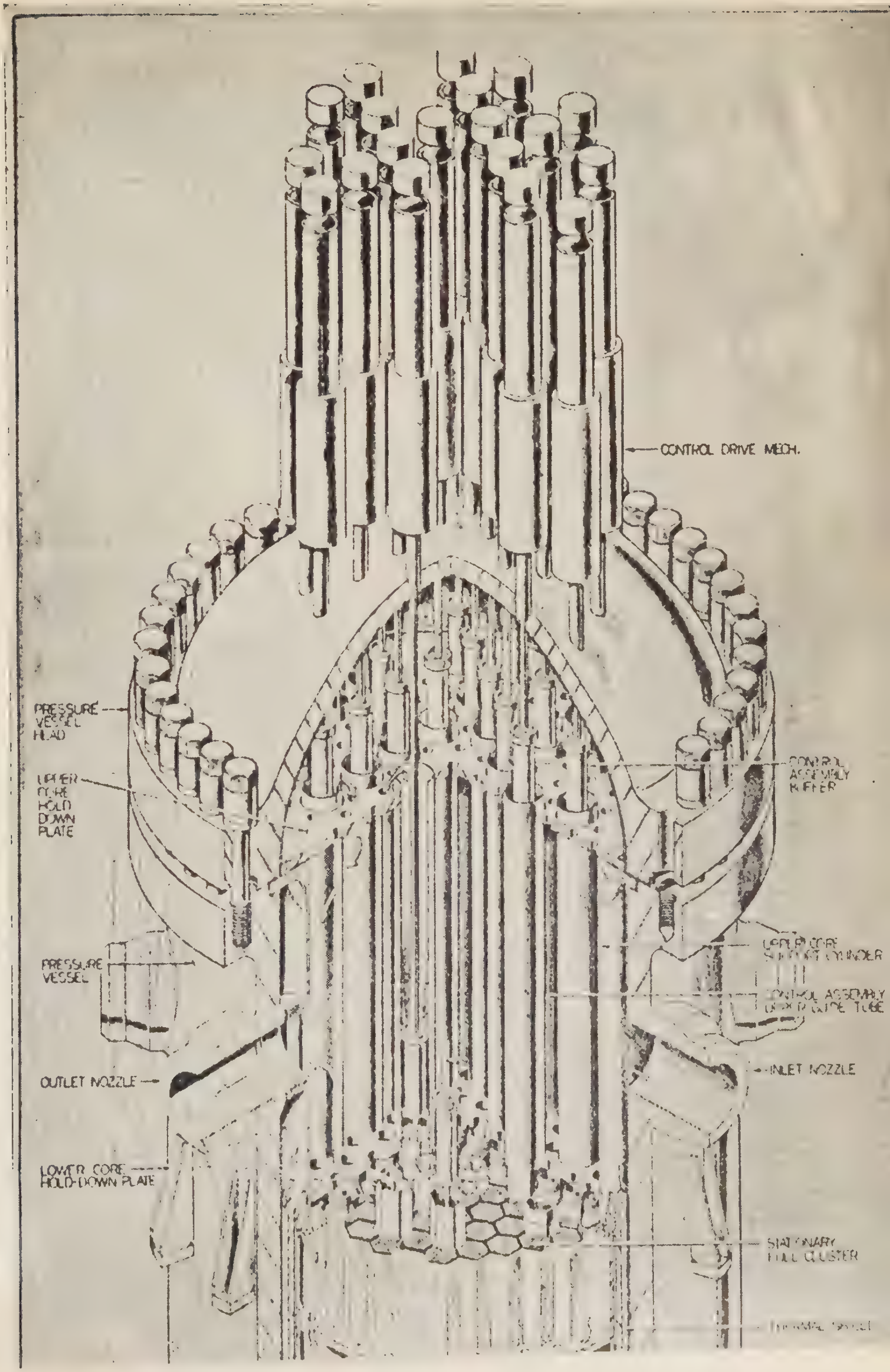
Figure. 1. Vertical section through advanced pressurized water reactor (all dimensions in inches)





LOWER CORE
HOLD-DOWN PLATE
PRESSURE VESSEL
STATIONARY FUEL
CLUSTER
THERMAL SHIELD
CORE SHROUD
CORE SUPPORT
CYLINDER
UPPER CORE
SUPPORT PLATE
LOWER CORE
SUPPORT PLATE
DIFFUSER
CONTROL ASSEMBLY
LOWER GUIDE TUBE
FUEL ZONE 114
$452\frac{1}{4}$
$382\frac{7}{8}$
66
126
9
144

Figure 2. Perspective view of advanced P.W.R.



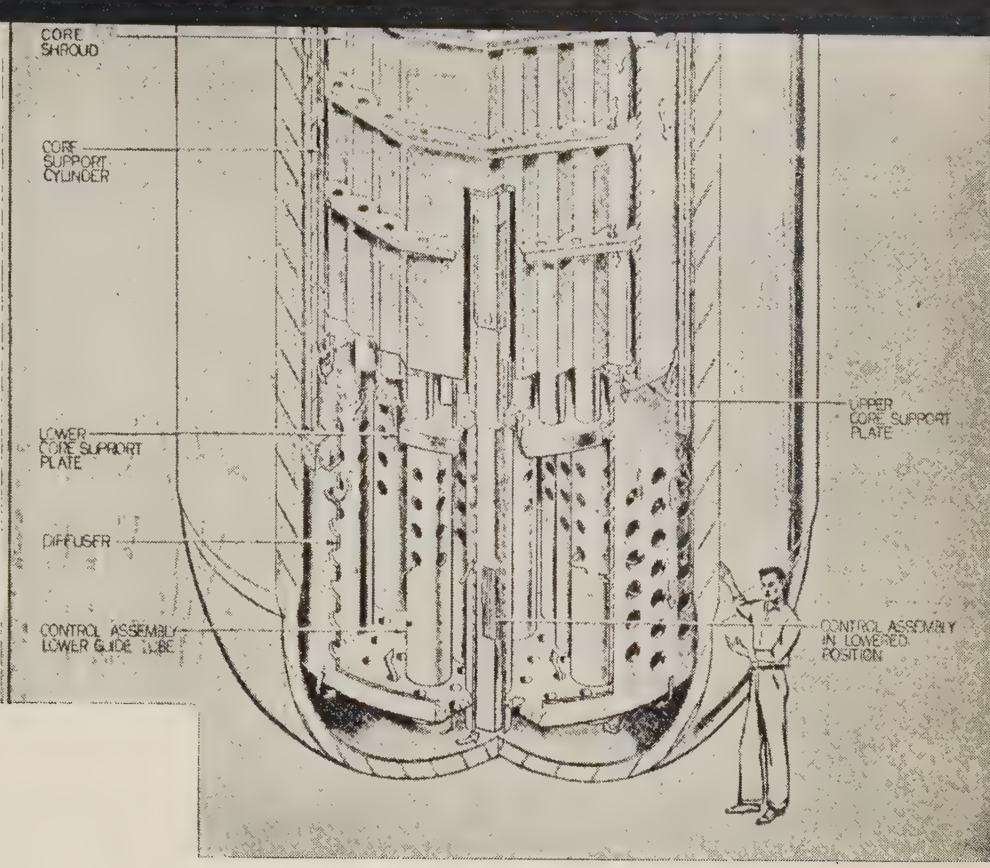
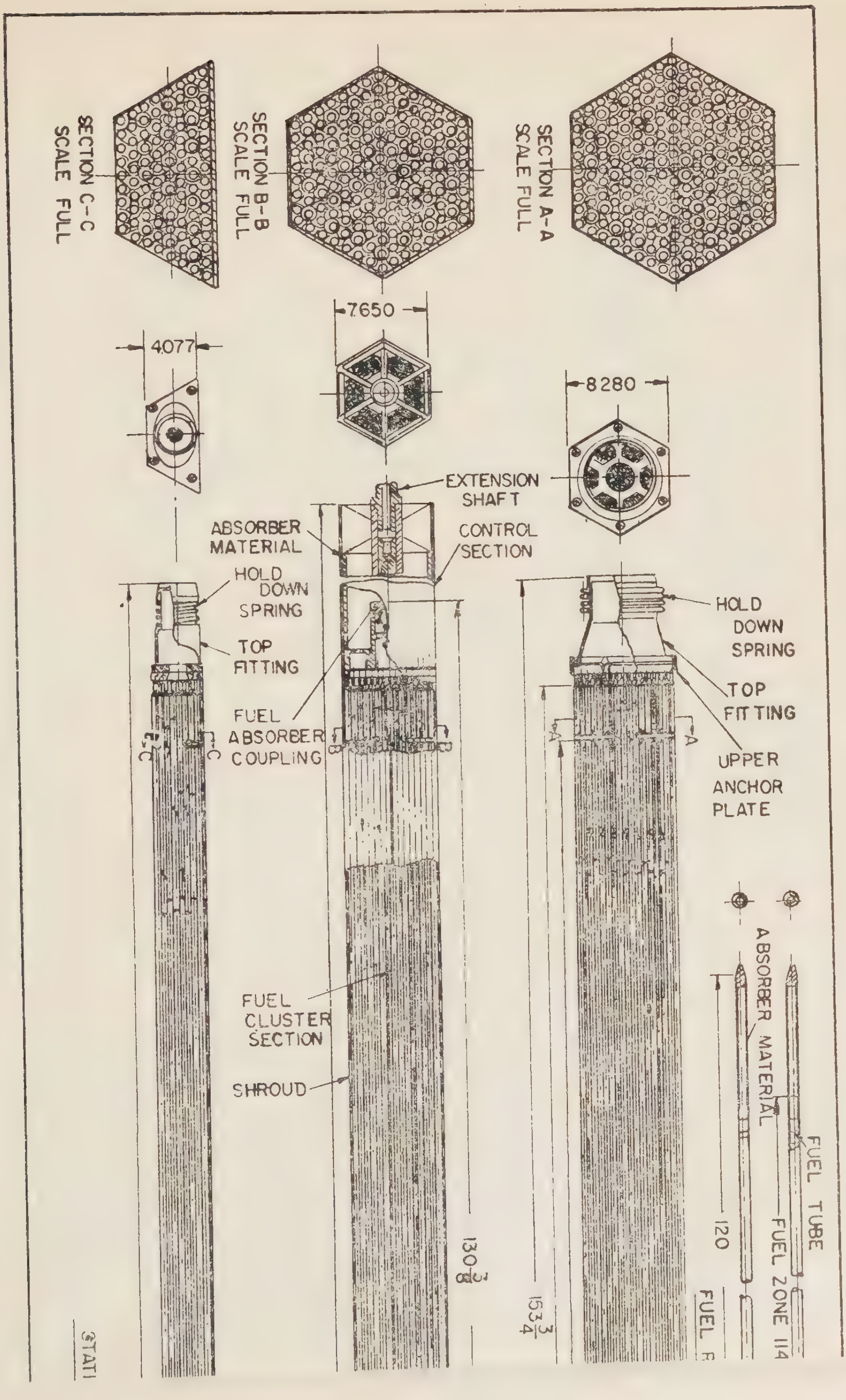

CORE
SHROUD
CORE
SUPPORT
CYLINDER
UPPER
CORE SUPPORT
PLATE
LOWER
CORE SUPPORT
PLATE
DIFFUSER
CONTROL ASSEMBLY
LOWER GUIDE TUBE
CONTROL ASSEMBLY
IN LOWERED
POSITION

Figure 3. Fuel cluster and control assembly for advanced P.W.R.



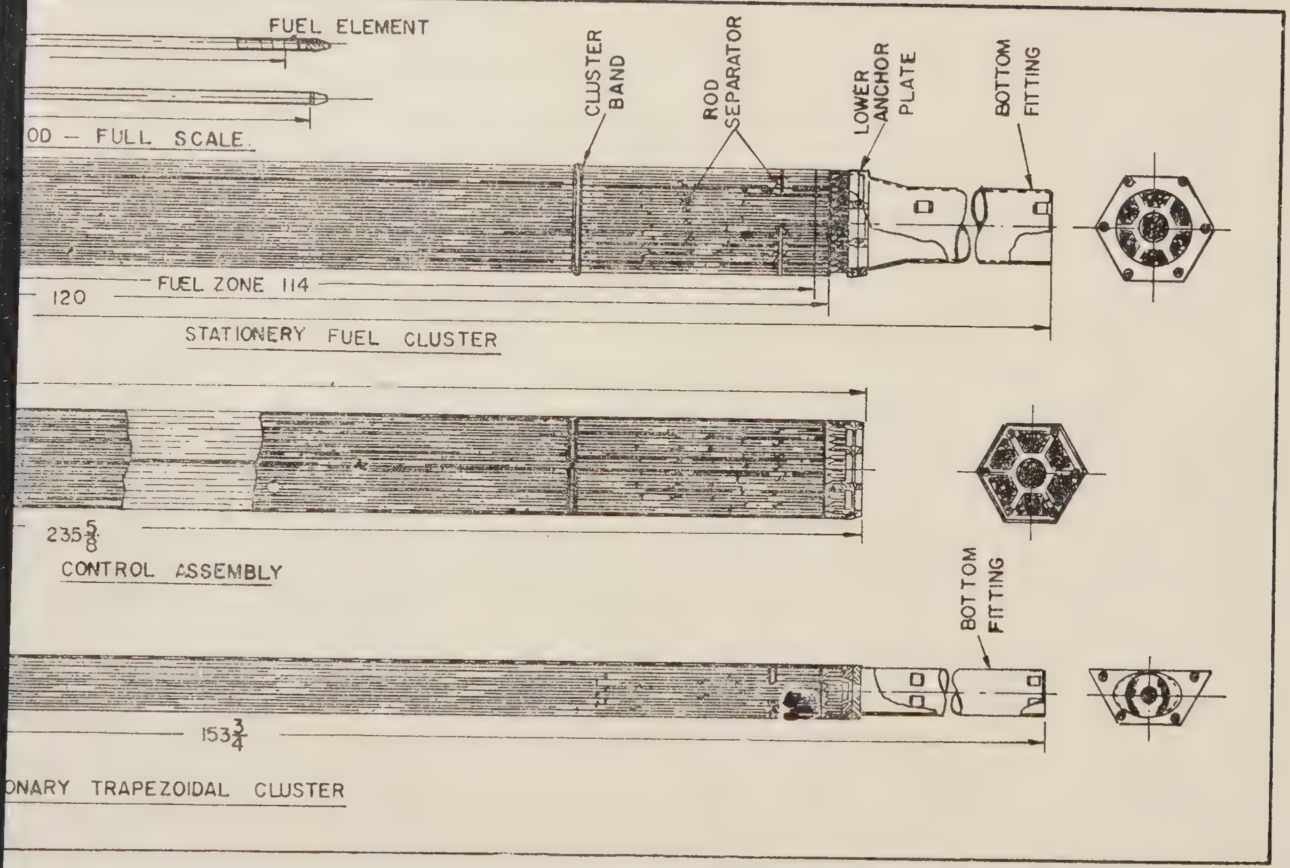
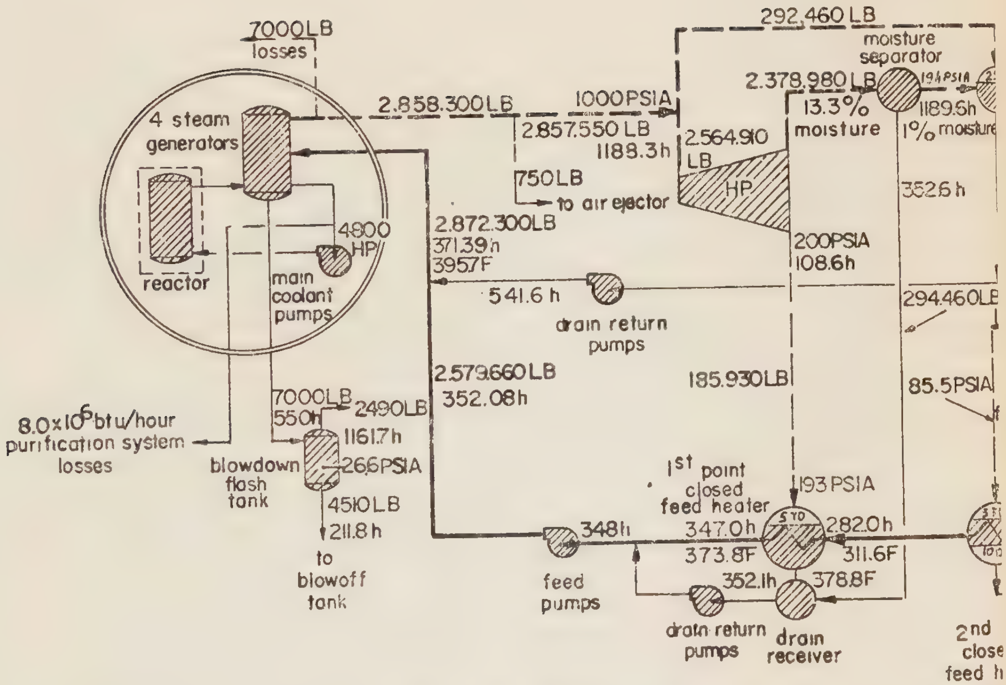

FUEL ELEMENT
OD – FULL SCALE.
CLUSTER BAND
ROD SEPARATOR
LOWER ANCHOR PLATE
BOTTOM FITTING
FUEL ZONE 114
120
STATIONERY FUEL CLUSTER
235 5/8
CONTROL ASSEMBLY
BOTTOM FITTING
153 3/4
ONARY TRAPEZOIDAL CLUSTER

Figure 5. Heat balance d



$$\text{Turbine heat rate} = \frac{(246.120+2540\pm3040-2040-2600)3412.75+1663.990(953.5-59.7)+124.170(7}{246.120}$$

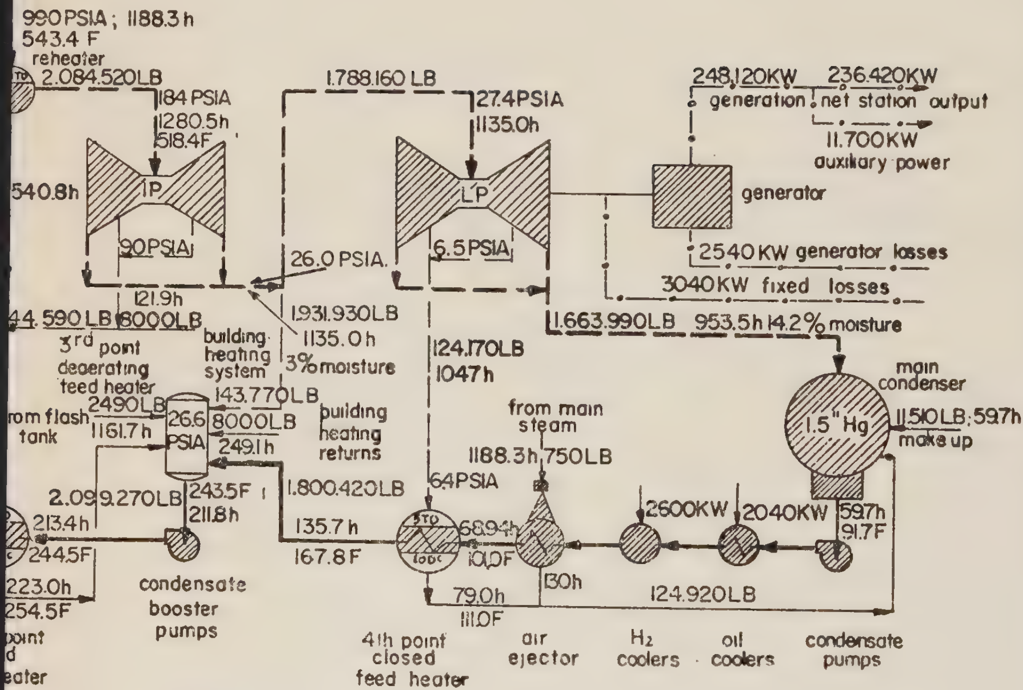
Purification system losses — 8 × 10; at all loads
Steam losses — 7000 LB at all loads
Steam generator 0.25% of steam generation
Extraction line pressure drops are expressed as the following percentage of the absolute pressre at the turbine connection to the extraction line

line	1st point	2nd point	3rd point	4th point
per cent	3.5	5.0	5.0....	1.0

Extraction pressure shown at or near turbine are pressures at tne turbine connection to the extractione line

Heater entrance pressures a symbols
The per cent pressure drop turbine exhaust to the rehea heater inlet to reheater out.
Auxiliary power requiremer from main coolant pump feed pumps and condens estimated for all other equ

iagram — advanced P.W.R.



$\overline{9.0-59.7}) = 9430 \dfrac{\text{BTU}}{\text{KWH}}$

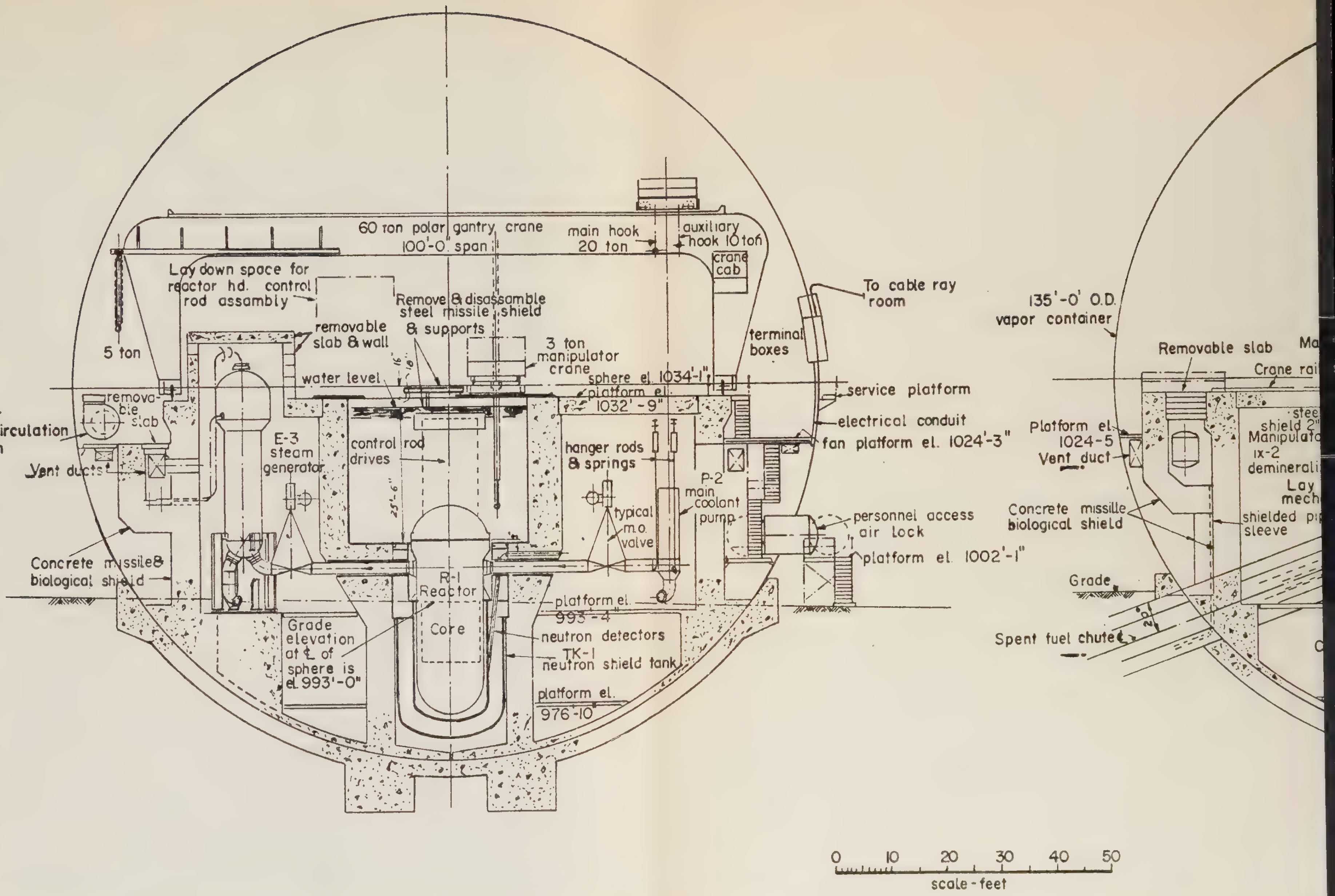
Station heat rate = 9890 BTU/KWH

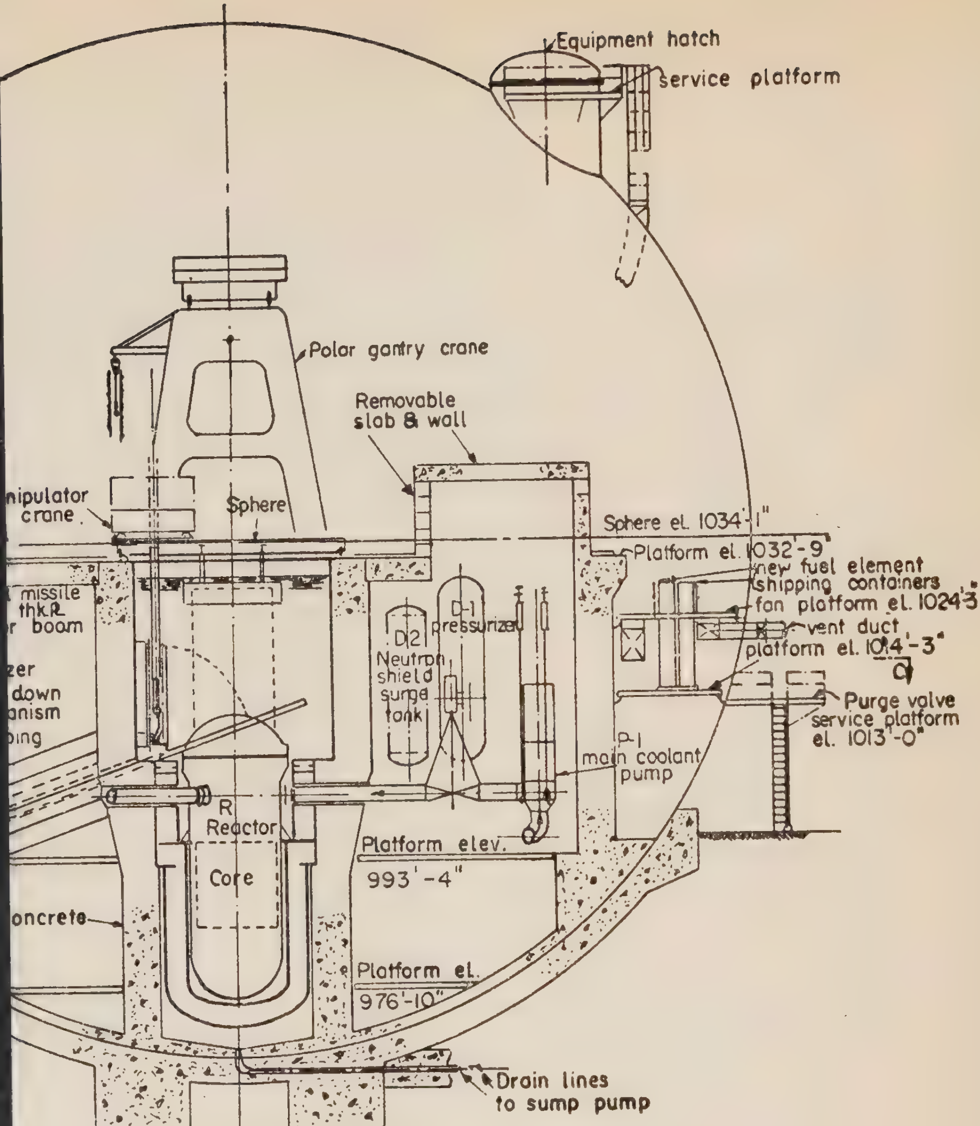
— — — — —	steam
—o—o—o—	water
————	power
LB	flow, pounds per hour
h	enthalpy BTU per pound
F	temperature, degrees fahr.
TD	terminal difference, degrees fahr.
DC	term. diff. drain cooler, degrees fa.
KW	kilowatts
"Hg	pressure, inches of mercury, absolute
PSIA	pressure, pound per square inch, absolute
PSIG	pressure, pound per square inch, gauge

re shown outside heater

from the high pressure
ter inlet and from the re-
et calculated at all loads
ts have been calculated
s, steam generator feed
te booster pumps, and
ipment.

Figure 6. General arrangement of advanced P.W.R. sections through vapor container





Equipment hatch
service platform
Polar gantry crane
Removable slab & wall
nipulator crane
Sphere
Sphere el. 1034'-1"
Platform el. 1032'-9"
new fuel element shipping containers
fan platform el. 1024'3
vent duct
platform el. 1014'-3"
missile thk.
r boom
zer down anism bing
D-1 pressurizer
D2 Neutron shield surge tank
Purge valve service platform el. 1013'-0"
P-1 main coolant pump
R Reactor
Core
Platform elev. 993'-4"
oncrete
Platform el. 976'-10"
Drain lines to sump pump

Figure 7. Possible boiling water reactor cycles

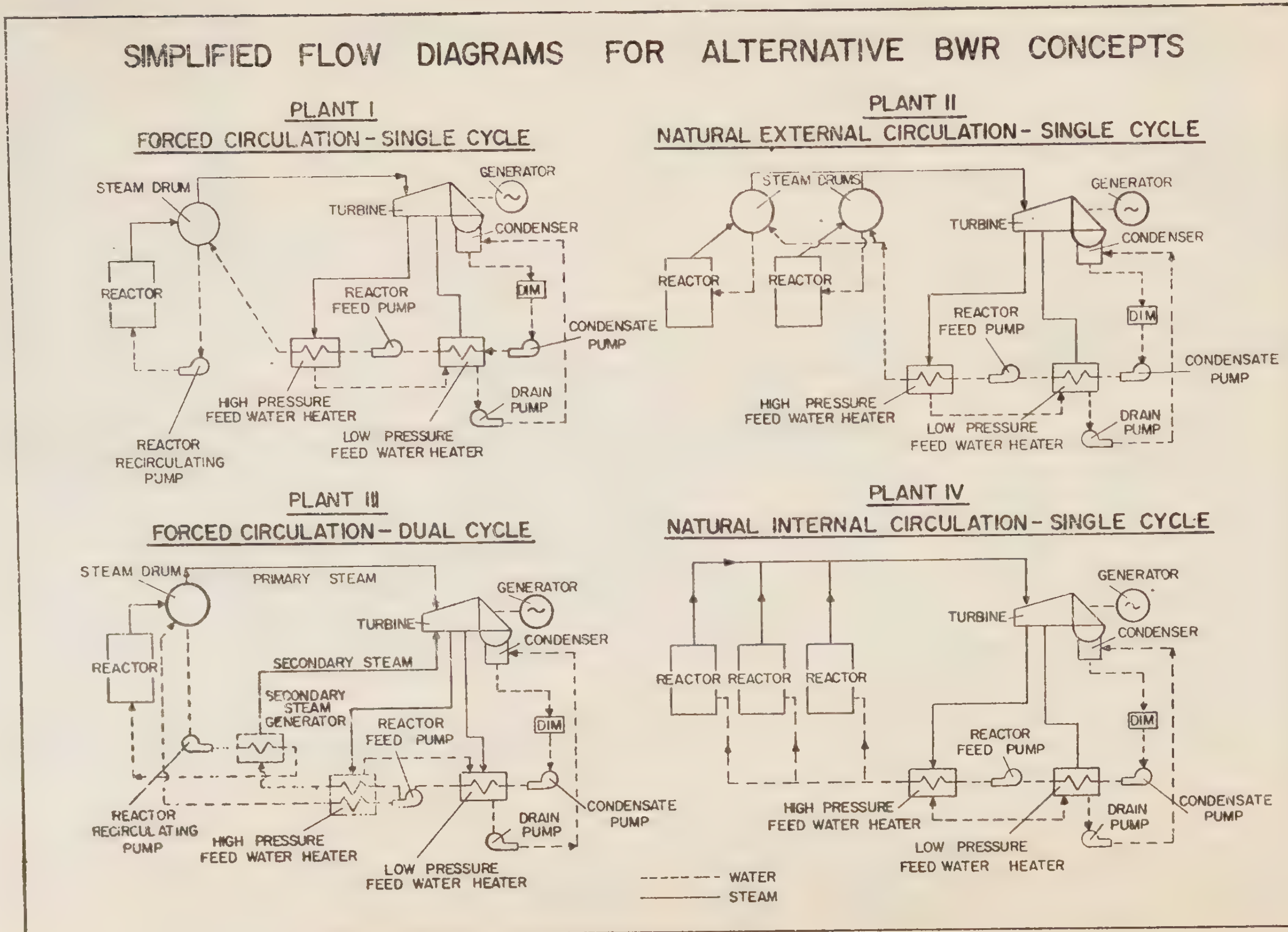


Figure 8. Flow diagram of advanced B.W.R.

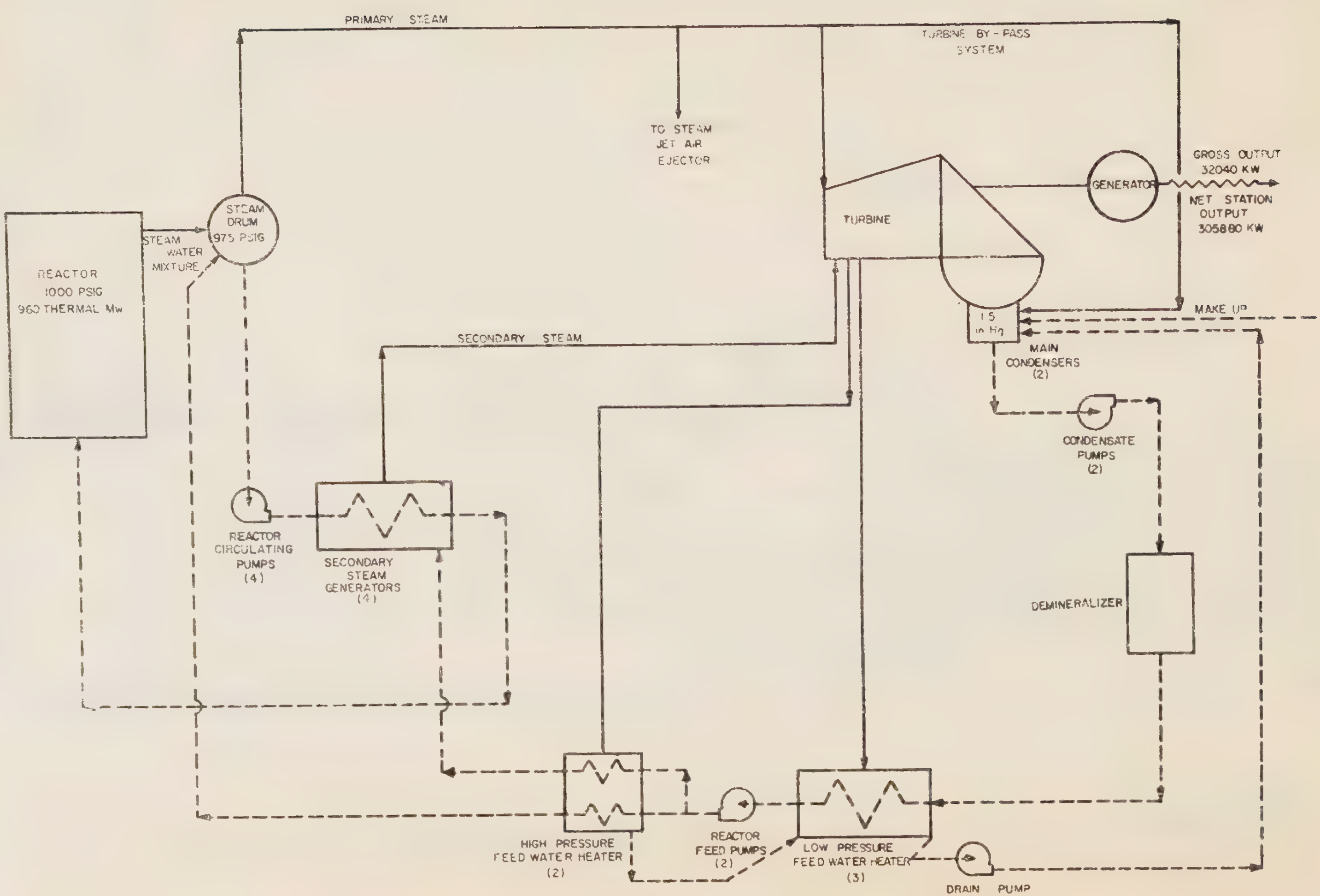


Figure 9. Section through pressure vessel — advanced B.W.R.

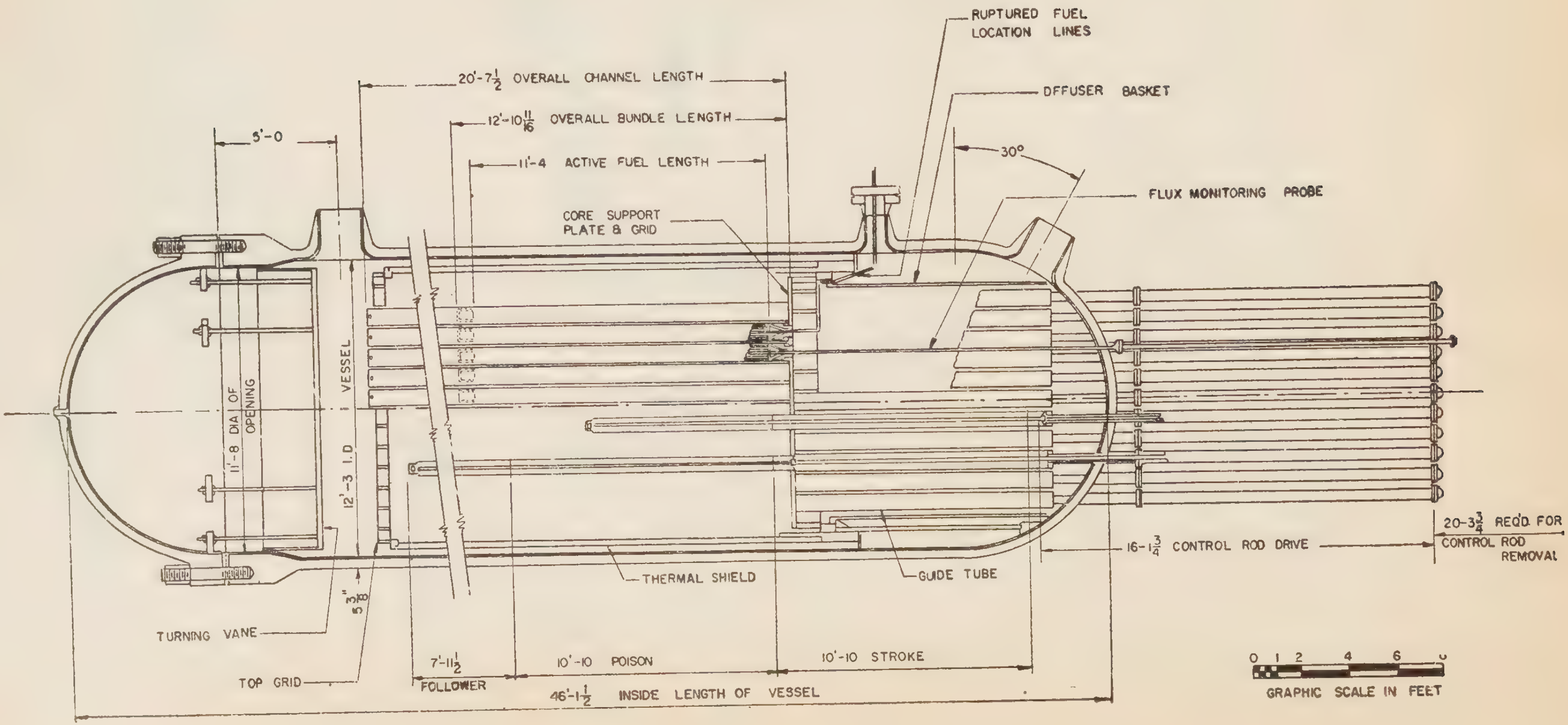
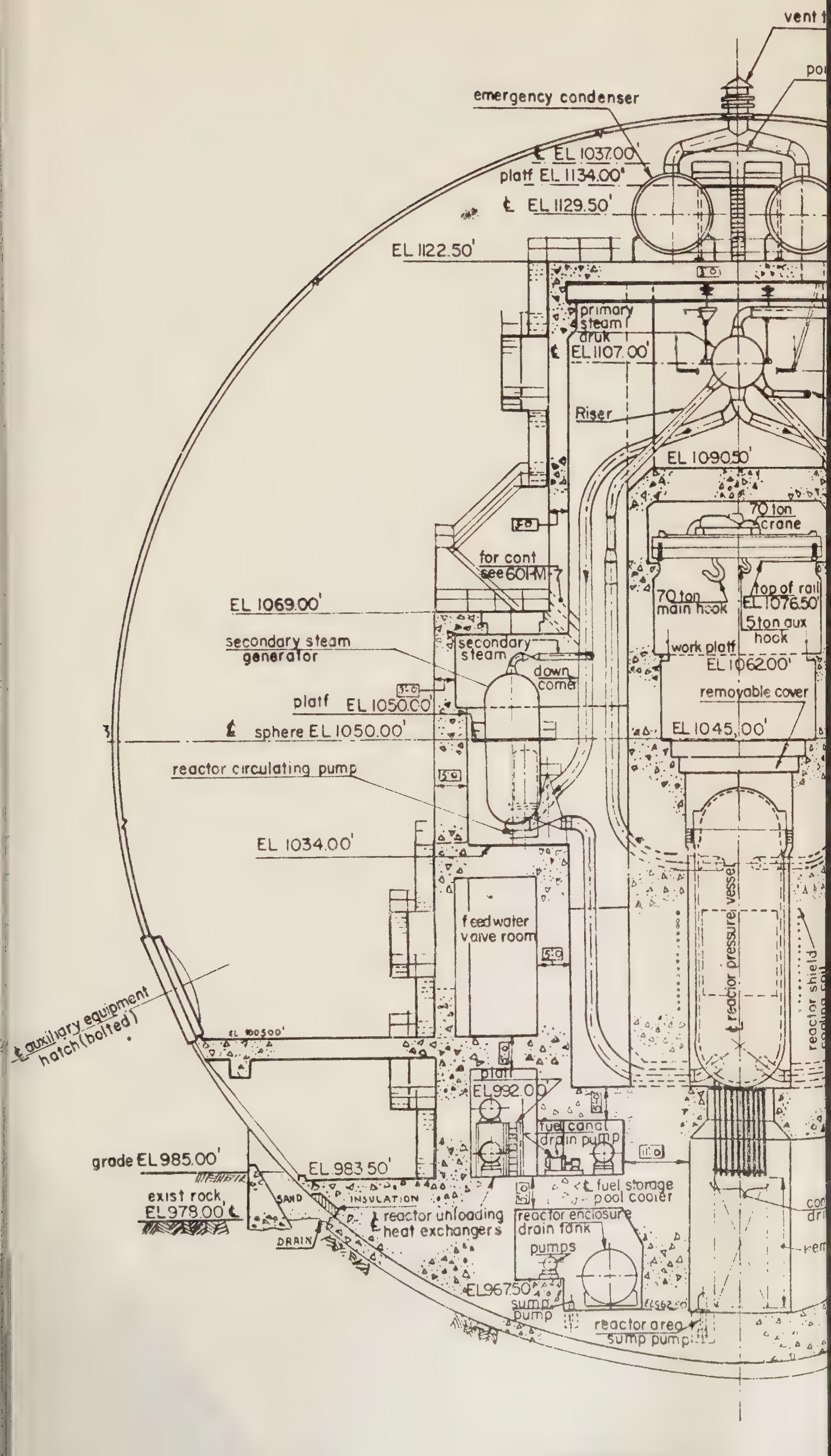
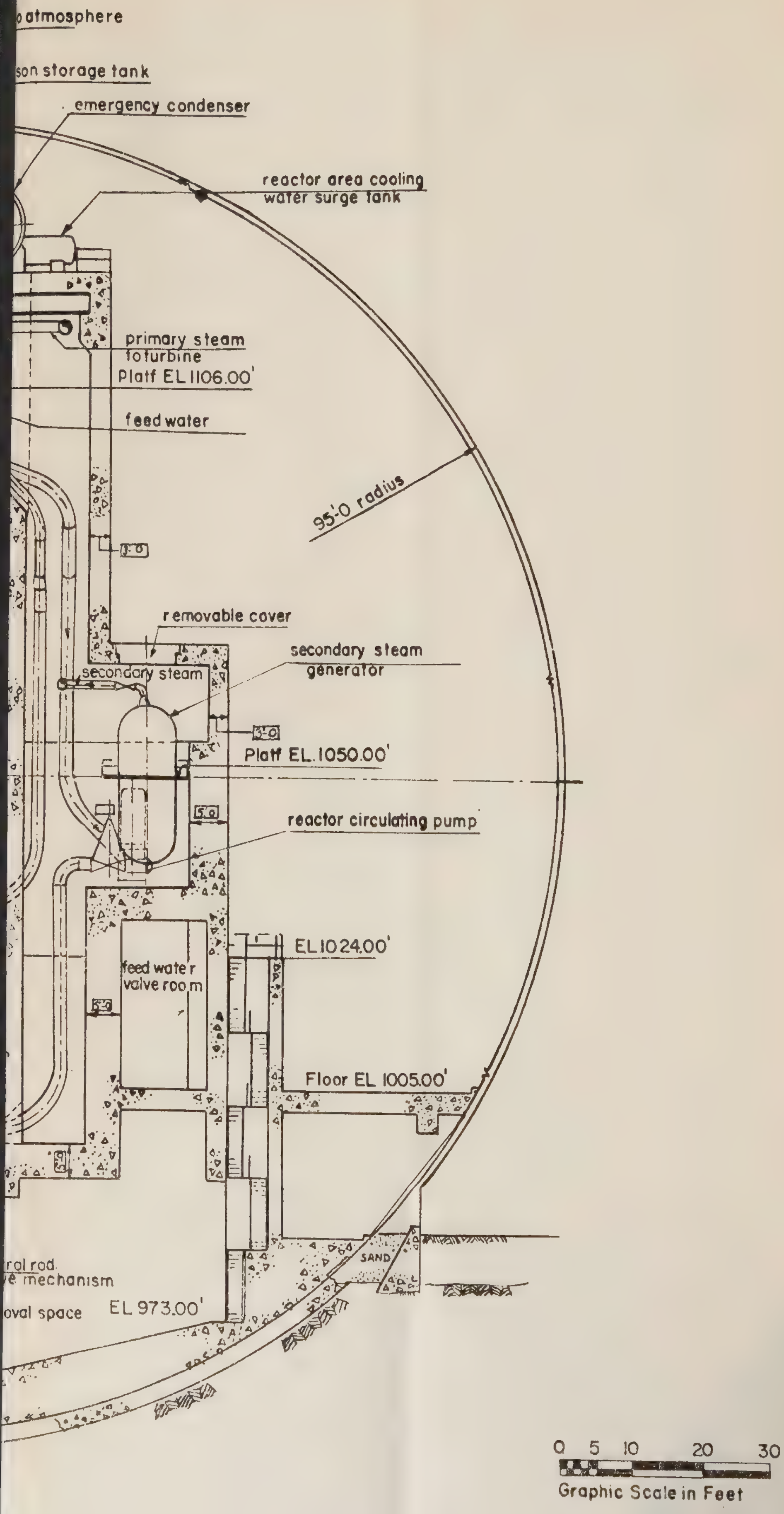


Figure 10. General arrangemen



of advanced B.W.R.



Fuel elements and control rods
advanced B.W.R.

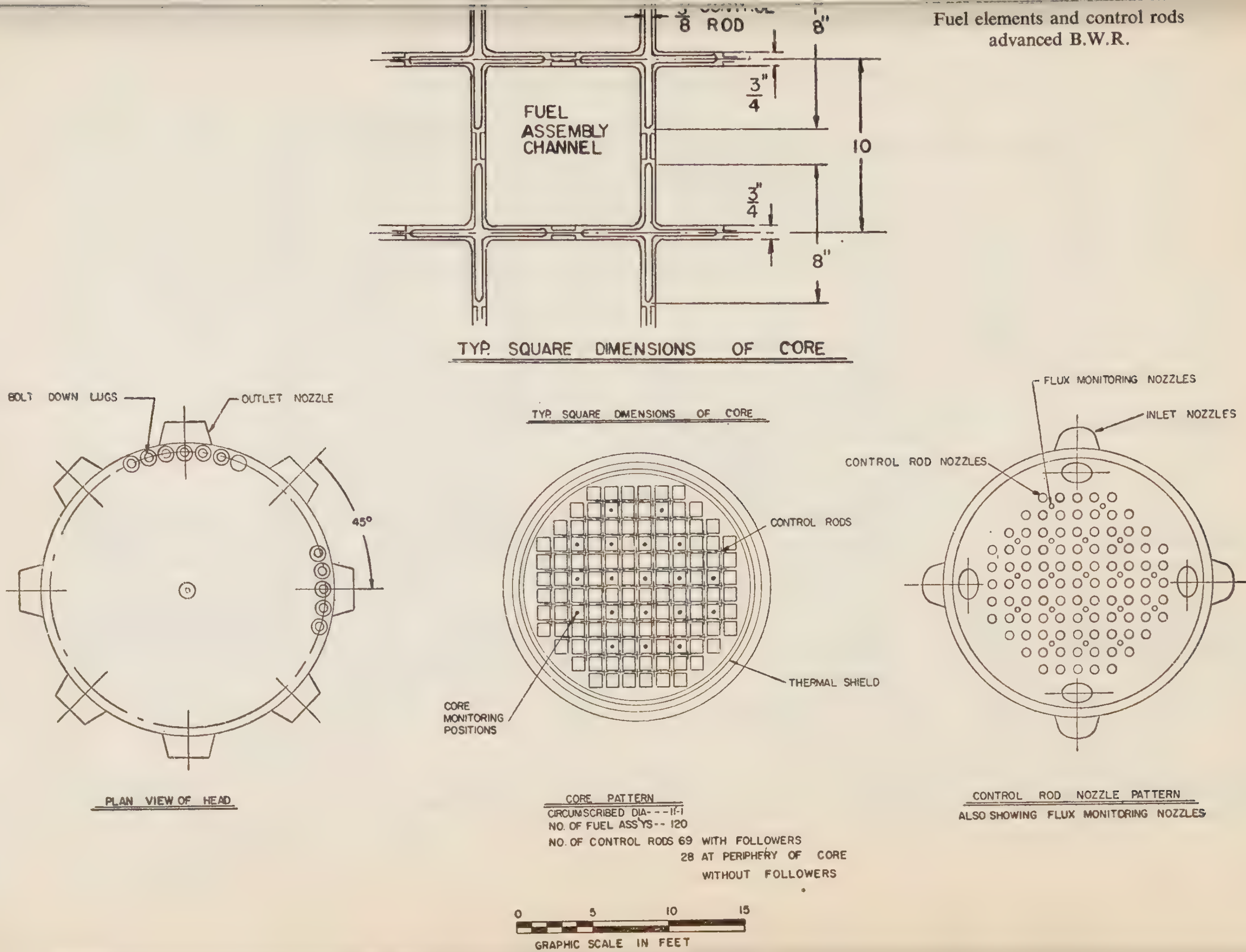
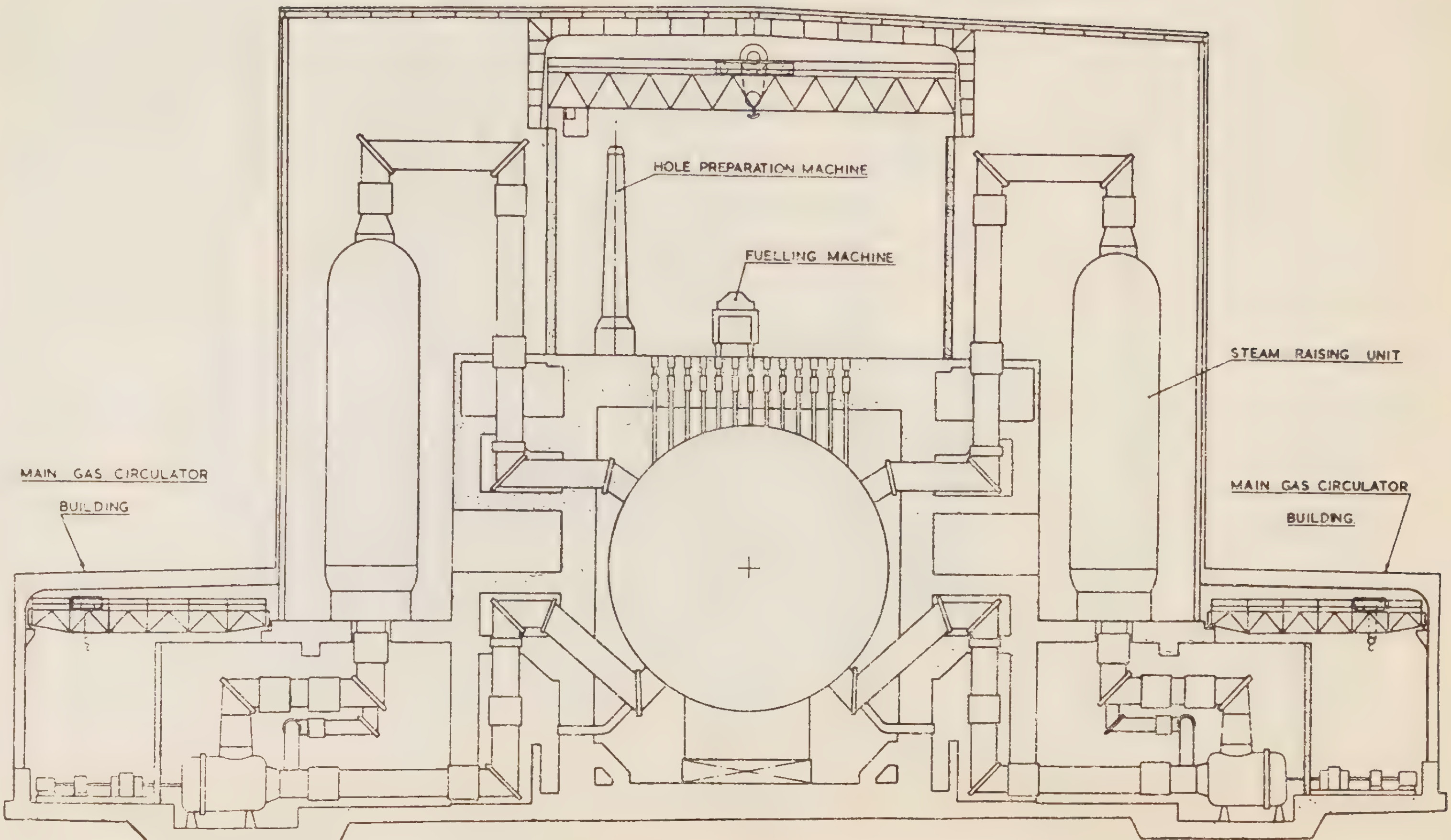


Figure 12. Section through Hinkley Point reactor station



Reactor No. 1 cross section

Figure 13. Graphite pile of Hinkley Point reactor

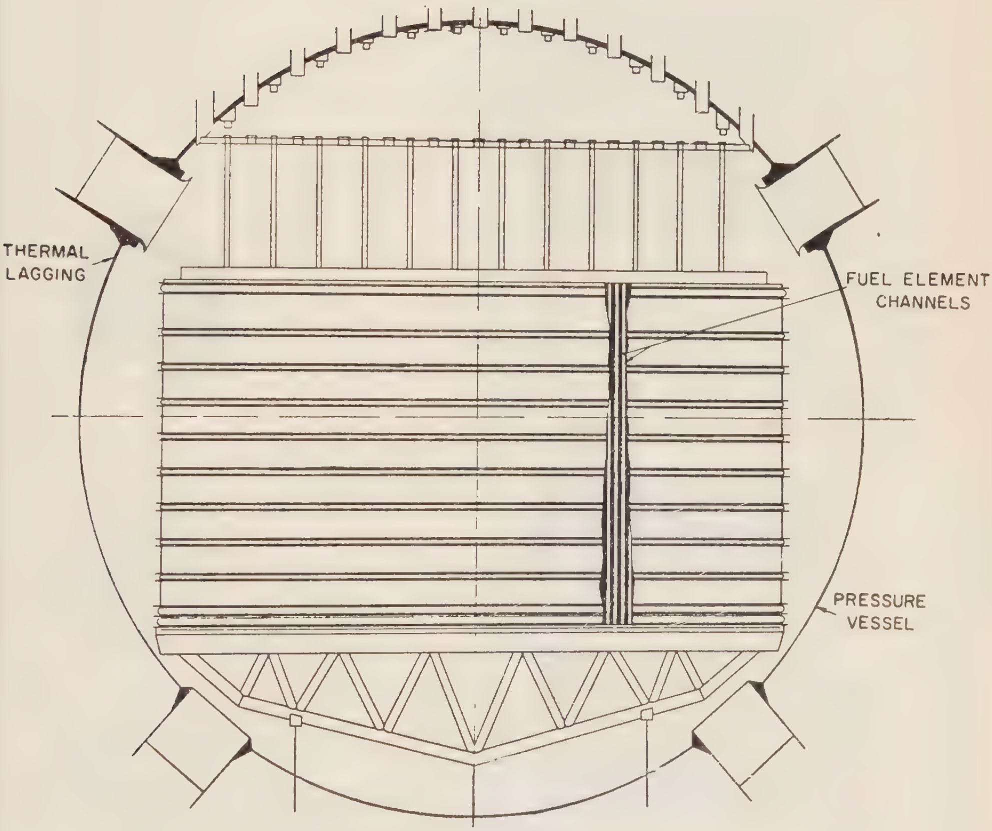
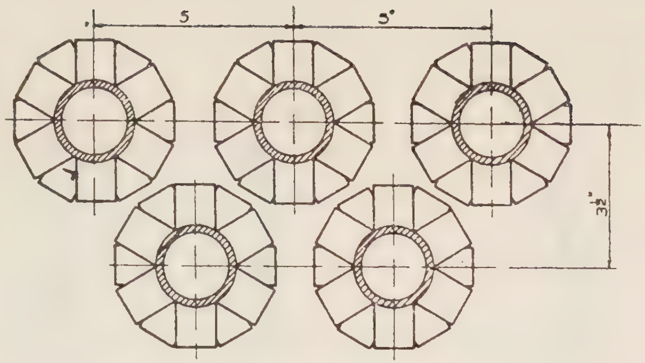
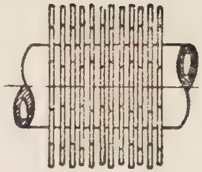
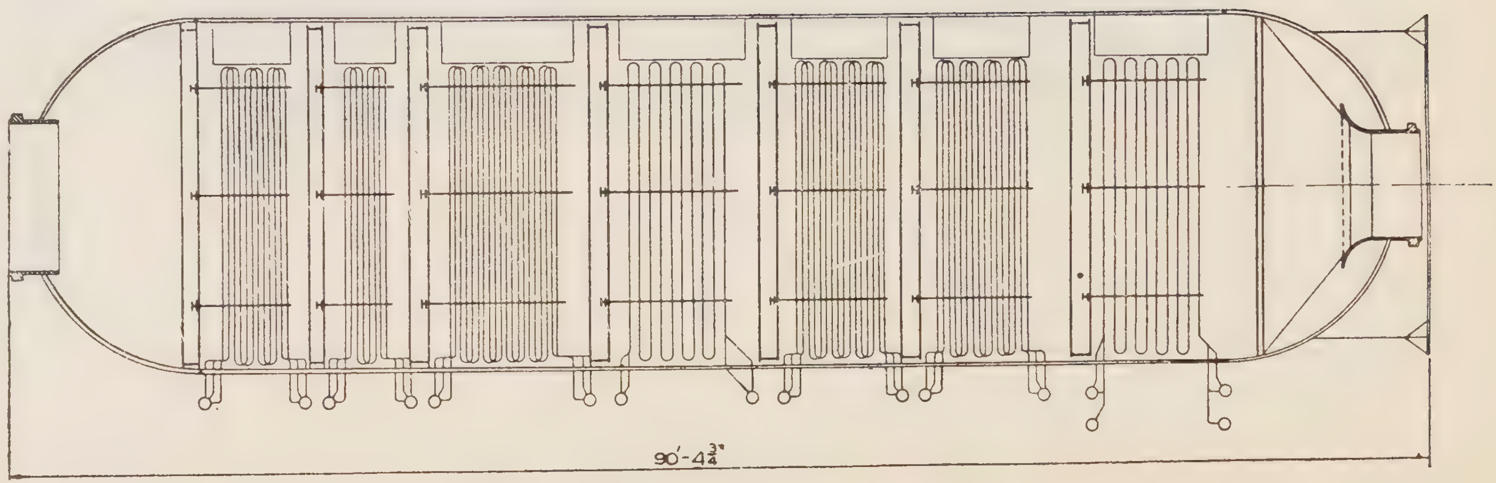


Figure 14.
Steam raisisng unit of Hinkley Point reactor



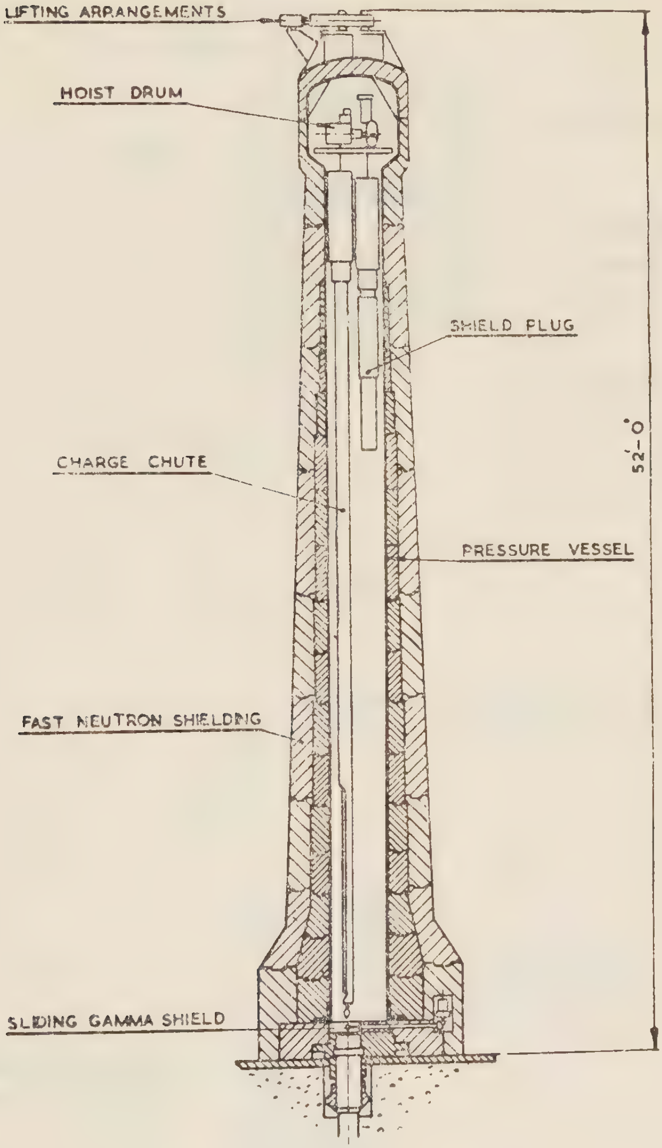
TYPICAL CROSS SECTION OF TUBES IN ALL BANKS





SECTION OF STEAM RAISING UNIT

Figure 15a.
Fuel handling machine of Hinkley Point reactor



HOLE PREPARATION MACHINE

Figure 15b.
Fuel handling machine of Hinkley Point reactor

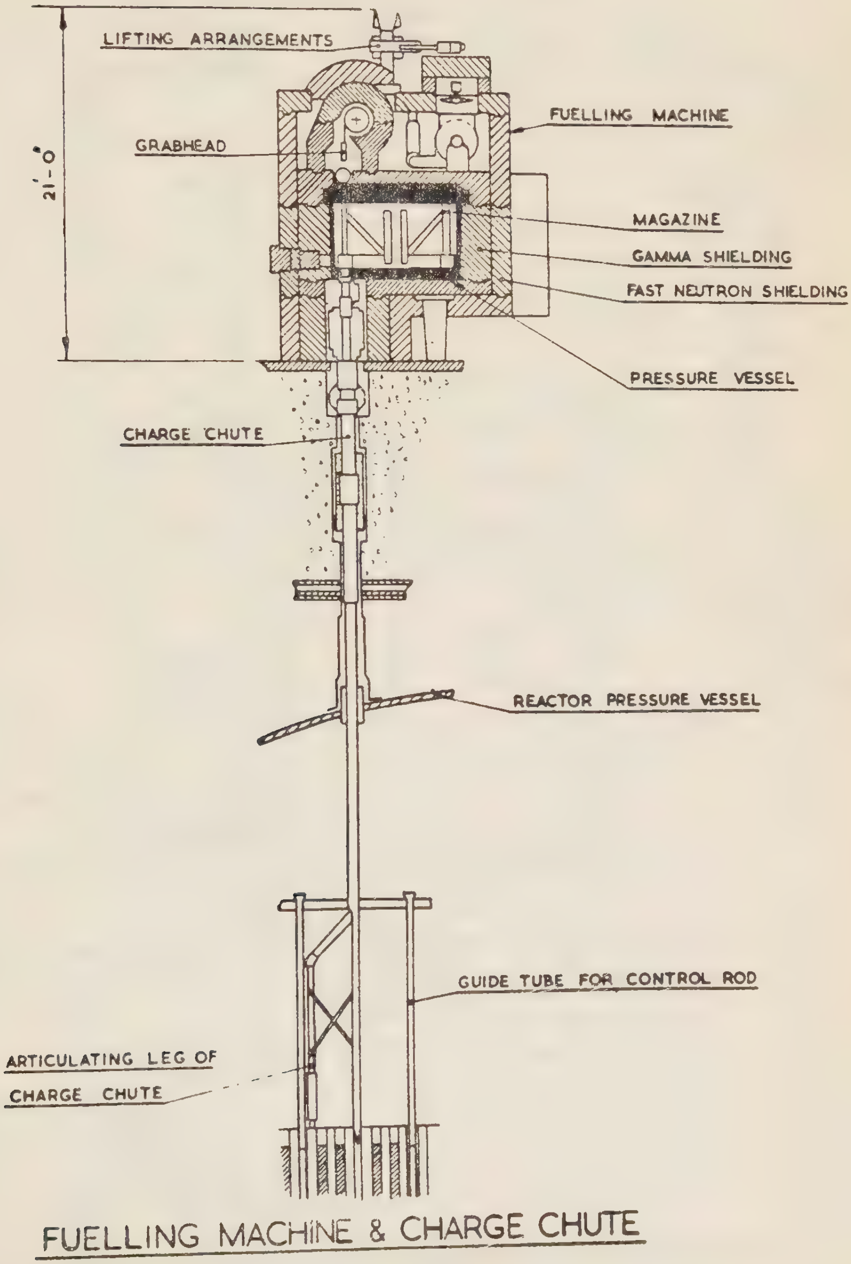
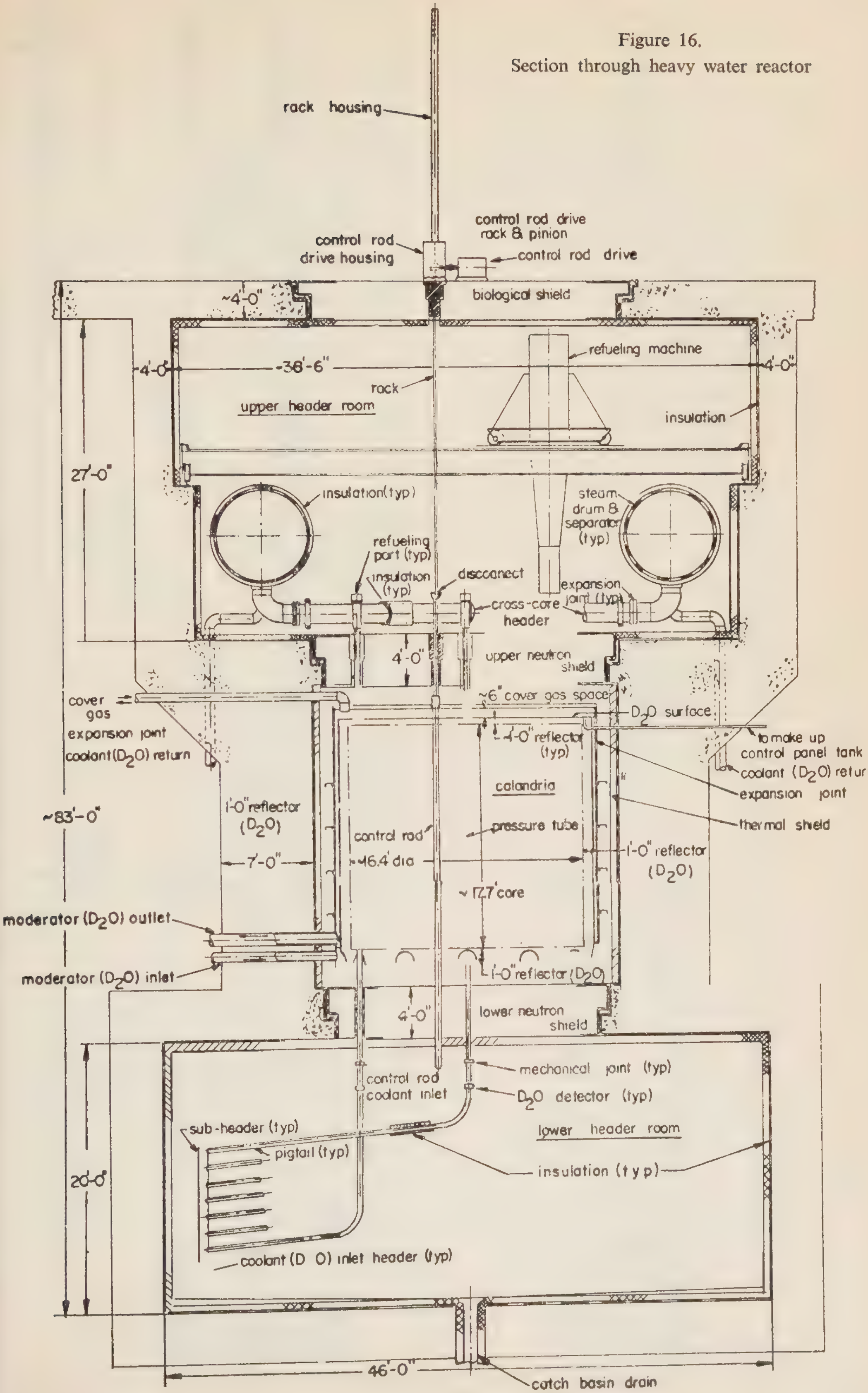


Figure 16.
Section through heavy water reactor



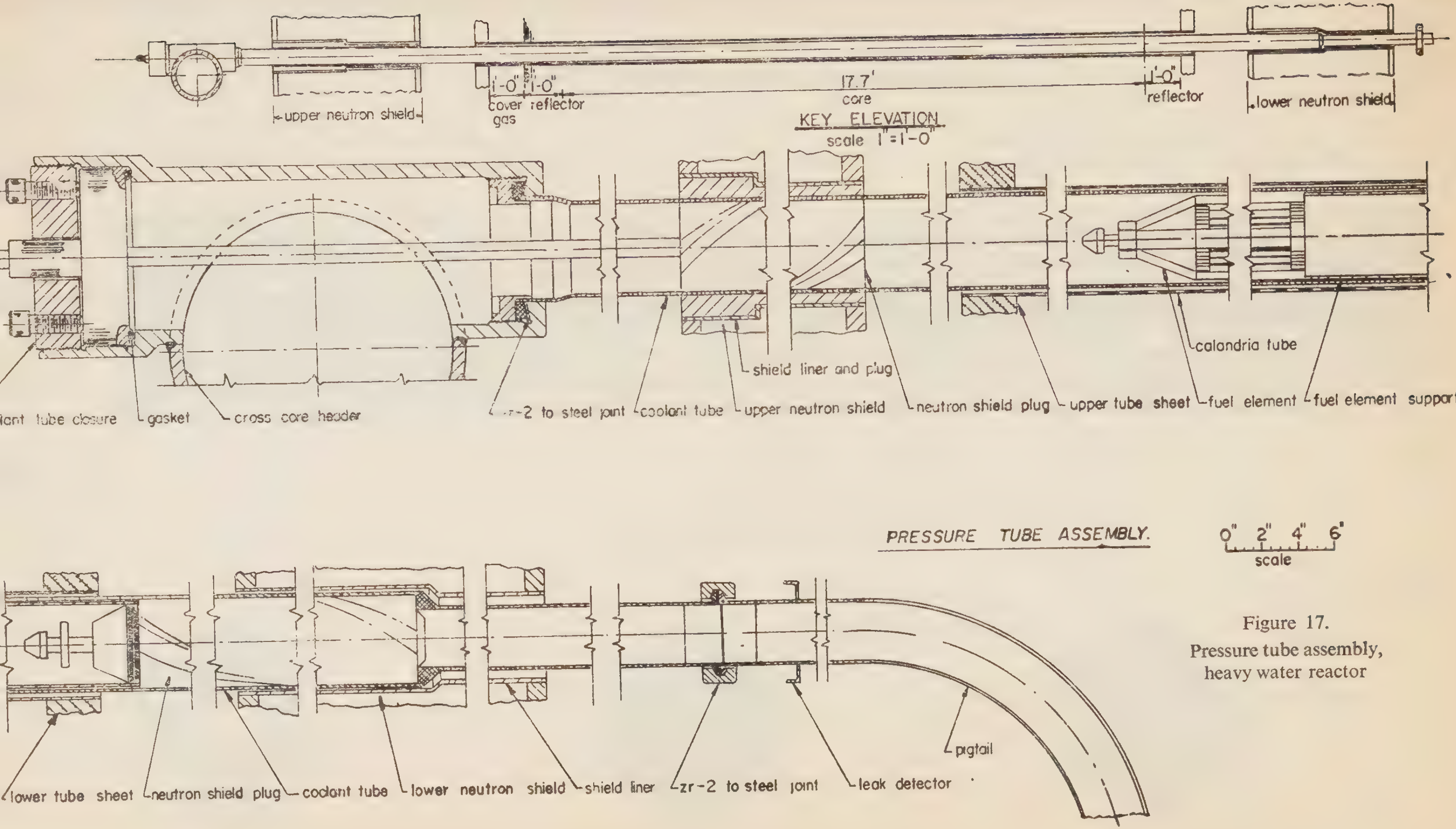
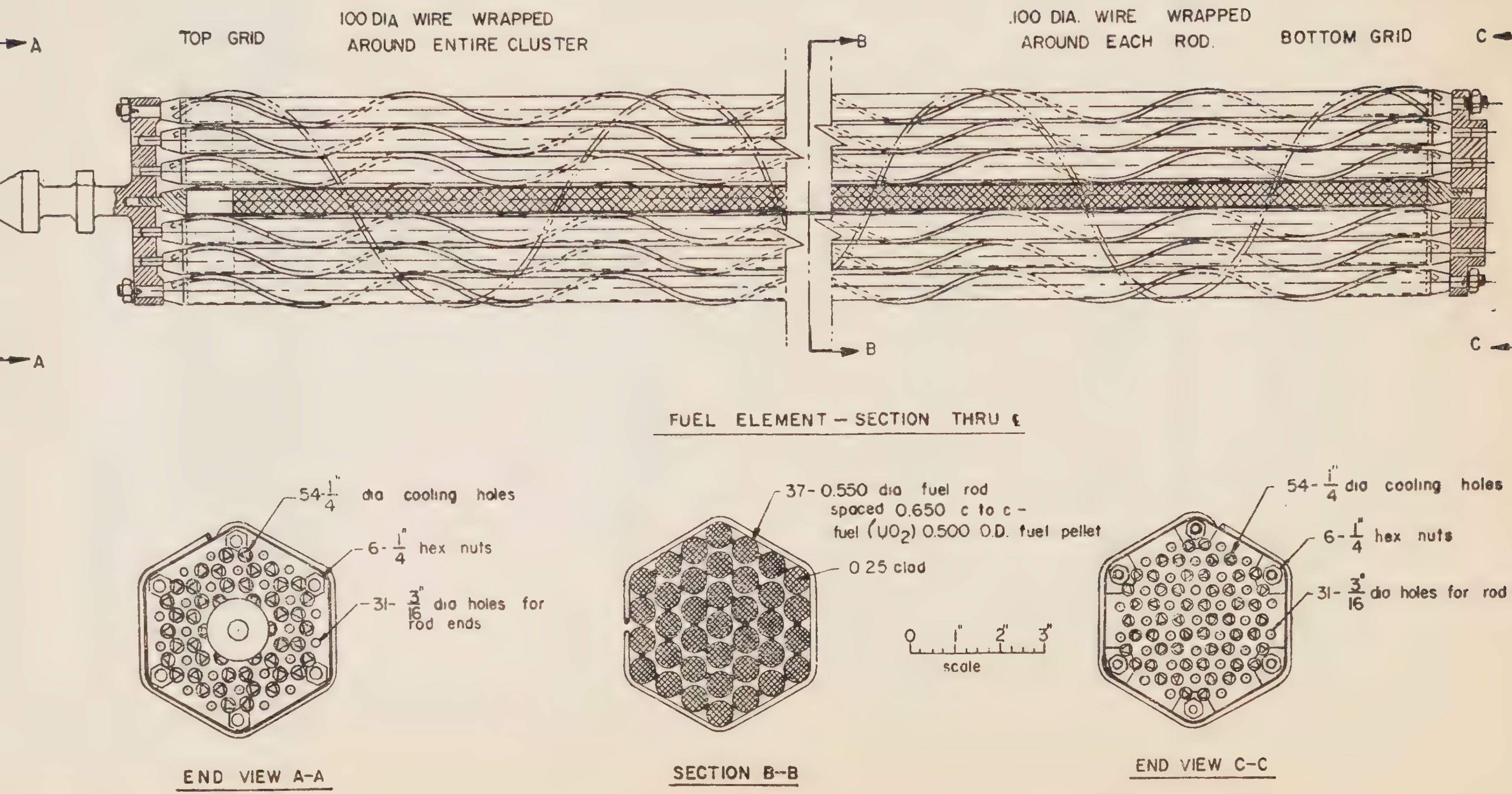


Figure 17.
Pressure tube assembly,
heavy water reactor

Figure 18. Fuel element assembly, heavy water reactor



LETTER TO THE EDITOR

About the critical wave lengths of perturbation in Bénard's problem

LIONEL RINTEL
Division of Mechanics, Technion—Israel Institute of Technology, Haifa

The equations

$$\begin{vmatrix} a_{11} & a_{13} & a_{15} & \cdots \\ a_{31} & a_{33} & a_{35} & \cdots \\ a_{51} & a_{53} & a_{55} & \cdots \\ \vdots & \vdots & \vdots & \\ \vdots & \vdots & \vdots & \end{vmatrix} = 0 \qquad (1)$$

$$\begin{vmatrix} a_{22} & a_{24} & a_{26} & \cdots \\ a_{42} & a_{44} & a_{46} & \cdots \\ a_{62} & a_{64} & a_{66} & \cdots \\ \vdots & \vdots & \vdots & \\ \vdots & \vdots & \vdots & \end{vmatrix} = 0 \qquad (2)$$

where

$$a_{mn} = (1 - R_a k^2/\lambda_n^3)\delta_{mn} + \frac{16k^3\pi^2{}_{mn}}{\lambda_n^3\ \lambda_m^2} R_a b_{mn}$$

$$\delta_{mn} = \begin{vmatrix} 1 & m=n \\ 0 & m\neq n \end{vmatrix}, \lambda_n = k^2 + n^2\pi^2$$

$$b_{mn} = \begin{vmatrix} \dfrac{\cosh^2 k/2}{\sinh k + k} & m,n \text{ odd} \\ \dfrac{\sinh^2 k/2}{\sinh k - k} & m,n \text{ even} \end{vmatrix}$$

are respectively the equations of the neutral stability curves for symmetric and anti-symmetric perturbations of the conductive heat flow in a horizontal layer of viscous fluid heated from below (Bénard's problem)[1,2]. When plotting R_a as a function of k these equations represent a series of curves convex downwards, each of which gives the neutral stability curve for a secondary mode of convective motion. The

Received June 15, 1960.

minimal values of R_a are the critical Rayleigh numbers $R_{a(cr)}$ for the specific mode and the corresponding values of k are the critical wave lengths of perturbation $k_{(cr)}$.

Numerical calculations made by approximating the infinite determinants (1) and (2) with finite determinants, and by solving the so obtained algebraic equations for different values of k, gave the following values for $R_{a(cr)}$ and $k_{(cr)}$:

mode	I	II	III	IV	V	VI
$R_{a(cr)}$	1.708×10^3	1.761×10^4	7.667×10^4	2.204×10^5	5.012×10^5	1.040×10^6
$k_{(cr)}$	3.117	5.365	7.613	9.862	12.110	14.358

It can be seen that $k_{(cr)}$ form an arithmetic progression with difference 2.248... I believe that this relationship is an exact one and valid also for the higher modes. Therefore, a deductive proof of it will be of interest and could lead to an understanding of the mathematical structure of the equations (1) and (2), and possibly to a better physical understanding of the phenomena of secondary flows.

ACKNOWLEDGEMENT

I wish to thank Prof. P. Lieber of the University of California who suggested the carrying out the calculations reported in this letter.

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Published by
THE WEIZMANN SCIENCE PRESS OF ISRAEL
Research Council of Israel, Ministry of Education and Culture
The Hebrew University of Jerusalem, Technion–Israel Institute of Technology
The Weizmann Institute of Science, Bialik Institute

Printed in Israel
JERUSALEM ACADEMIC PRESS LTD.
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WSP/1000/7.60/46